ABB

ASEA BROWN BOVERI

April 30, 1993
LD-93-072

Docket 52-002

Attention: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SYSTEM 80+ Supplemental Information

Dear Sirs:

Attached is information requested by the staff to supplement information on the System 80+ design descriptions which is already on the docket.

ABB-CE has initiated an Integrated Review of the CESSAR-DC and Design Descriptions/ITAAC to ensure consistency among and within these documents. It is possible that changes to the attached material may be necessary should the review uncover any inconsistencies. It is our intention to incorporate such changes in our final amendment targeted for June 30, 1993.

If you have questions related to this material, please contact me or Mr. John Rec (203-285-2861).

Very truly yours,

COMBUSTION ENGINEERING, INC.

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C. B. Brinkman
Acting Director
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For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR NUCLEAR ISLAND (NI) STRUCTURES (2.1.1)

1. Amplifying Information for the NI Structures

   See CESSAR-DC Sections 3.3 through 3.8, 6.2.1, and 6.2.6.

2. CESSAR-DC Chapter 14 Tests Applicable to the NI Structures

   See CESSAR-DC Section 14.2.12.1.130

REFERENCE INFORMATION FOR NUCLEAR ISLAND (NI) STRUCTURES (2.1.1)

Relationship of the Safety Analysis to the NI Structures

The Containment must withstand the pressure and temperatures of the DBA without exceeding the design leakage rate.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR NUCLEAR ISLAND (NI) STRUCTURES
(2.1.1)

Relationship of the PRA to the NI Structures

Physical separation is provided between redundant divisions of safety-related equipment.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR NUCLEAR ISLAND (NI) STRUCTURES (2.1.1)

Relationship of the Shutdown Risk Evaluation to the NI Structures

None

TABLE 3.2-1 (Cont'd)

(Sheet 23 of 27)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Structures				
Reactor Building				
Shield Building	2	I	RB	1
Steel Containment Vessel	2	I	RB	1
Internal Structure	3	I	RC	1
Equipment Hatch	2	I	RC	1
Personnel Airlocks	2	I	RC	1
Subsphere	3	I	RB	1
Nuclear Annex				
Control Area	3	I	NA	1
EFW Tank/Main Steam	3	I	NA	1
Valve House Area	3	I	NA	1
Emergency Diesel	3	I	NA	1
Generator Areas	3	I	NA	1
CVCS/Maintenance Area	3	I	NA	1
Spent Fuel Pool Area				
Unit Vent	3	I (See Note 30)		1
Turbine Building	NNS	II	TB	2
Radwaste Facility (28)	3	II	RW	2
Station Service Water Pump Structure	3	I	SP	1
Station Service Water Intake Structure	3	I	SI	1
Component Cooling Water Heater Exchanger Structure	3	I	CX	1
Diesel Fuel Storage Building	3	I	DF	1

TABLE 3.2-1 (Cont'd)

(Sheet 27a of 27)

CLASSIFICATION OF
STRUCTURES, SYSTEMS, AND COMPONENTS

NOTES: (Cont'd)

(29) The QA program provides a graded approach to the assurance of quality of work performed by and for ABB-CE by the use of quality class designations to describe the various levels of controls as follows:

1) QC-1 is the highest level quality class and embodies all necessary controls for items and/or services which are required to meet 10 CFR 50 Appendix B requirements.

2) QC-2 is an intermediate level quality class which is used for items or services which require a moderate level of control of activities affecting quality, but which are neither Nuclear Safety-Related nor required to meet the requirements of 10 CFR 50 Appendix B. Circumstances appropriate for QC-2 designation include non-standard, complex items, or those which must perform reliably, in a harsh environment or with less than normal operator attention or maintenance.

3) QC-3 is the quality class which applies to all items or services which are not assigned to another quality class. Quality requirements may be specified in quality plans, procurement documents and/or special procedures if deemed necessary.

(30) The Unit Vent is Seismic Category I and will not be designed for tornado wind and wind pressures or tornado generated missiles.

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These Seismic Category I ~~safety-related~~ structures are designed to ~~maintain a dry environment during all~~ protect safety-related equipment from floods by incorporating the following safeguards into their construction:

A. No exterior access openings will be lower than 1 foot above plant grade elevation.

B. The finished yard grade (yard grade) adjacent to the safety-related structures will be maintained at least 1 foot below the ground floor elevation, except where ramps or steps are provided for access.

C. Waterstops are used in all horizontal and vertical construction joints in all exterior walls up to flood level elevation.

D. Water seals are provided for all penetrations in exterior walls up to flood level elevation.

~~E. Waterproofing of walls subject to flooding is provided.~~

For other Seismic Category I ~~safety related~~ structures where flood protection measures are required (e.g. pumping systems, stoplogs, watertight doors, dikes, retaining walls and drainage systems) the design of means for providing such protection will be described in Section 2.4 of the site-specific SAR. Insert 1

Insert 2

~~Redundant equipment is separated and compartmentalized so that a single flooding event does not affect redundant safety systems. Equipment such as the auxiliary shutdown panels are elevated off the floor so that flooding events will not affect these important pieces of equipment.~~

3.4.5 ANALYTICAL AND TEST PROCEDURES

A description of the methods and test procedures by which static and dynamic effects of the design basis flood conditions or design basis groundwater conditions are applied is detailed in Section 2.4 of the site-specific SAR.

INSERT 1

Penetrations located below the external flood level in the external walls of the Nuclear Annex are currently projected to include Component Cooling Water, Radwaste, and Diesel Fuel Oil System piping and cable penetrations. Additional penetrations may be identified once layouts are finalized for systems such as sewage, demineralized water, station air, and security.

INSERT 2

External flooding is a result of secondary flooding sources located in the Turbine Building are addressed in Section 10.4.1.3. Entrances to the Nuclear Annex from the Turbine Building are elevated above plant grade to prevent flood propagation.

Internal Flood protection in the System 80+™ design minimizes possible flood sources. The station service water system is located outside the Nuclear Annex to eliminate a significant source of water. The component cooling water system and emergency feedwater are fully separated by division, thus eliminating the possibility of a single flood source within these systems impacting both divisions.

Lengths of high energy and moderate energy piping have been minimized by equipment location. Equipment in the RB Subsphere is located in quadrants to minimize the lengths of piping runs. The RB Subsphere provides for close proximity of equipment to reduce piping runs from containment.

Flood barriers have been integrated into the design to provide further flood protection while minimizing the impact on maintenance accessibility. The primary means of flood control in the Nuclear Annex and RB Subsphere is provided by the divisional wall which serves as a barrier between redundant trains of safe shutdown systems and components. Each half of the Subsphere is further divided into two quadrants to separate redundant safe shutdown components to the extent practical. Flood barriers provide separation between subsphere quadrants, while maintaining equipment removal capability. Emergency Feedwater pumps are located in separate compartments within the quadrants with each compartment protected by flood barriers.

Penetrations are sealed and no doors are provided up to EL.70+0, the maximum internal flood level in the divisional wall that separates the Nuclear Annex and the Reactor Building Subsphere. Where flood doors are provided, open and close sensors are also provided with status indication. Flood barriers also provide separation between electrical equipment and fluid mechanical systems at the lowest elevation within the Nuclear Annex. At higher elevations, safety-related electrical components are elevated above the floor so that flooding events will not affect components. Additional barriers (e.g., curbs, sealed penetrations) are provided or safety-related electrical components are elevated, as necessary, to mitigate the effects of postulated pipe rupture addressed in Section 3.6.

Flood protection is also integrated into the floor drainage system. The floor drainage systems are separated by division and Safety Class 3 valves are provided to prevent backflow of water to areas containing safety related equipment. Each subsphere quadrant is provided with redundant Safety Class 3 sump pumps and associated instrumentation, which are powered from the diesel generators in the event of loss of offsite power.

The Nuclear Annex floor drainage system is divisionally separated, with no common drain lines between divisions. Floors are gently sloped to allow good drainage to the divisional sumps.

Flood protection is incorporated into the Component Cooling Water Heat Exchanger Structure. This structure is divisionally separated by a wall such that a flood in one division can not flood the other division.

The Diesel Generator Building floor drain sump pumps and associated instrumentation are Safety Class 3 to prevent flooding of the diesel generators. These pumps are also powered from the diesel generator in the event of loss of offsite power.

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3.6.2.3.2.5 Design Criteria

The unique features in the design of pipe whip restraint components relative to the structural steel design are geared to the loads used and the allowable stresses. These are as follows: E

A. Energy-absorbing members are designed for the restraint reaction and the corresponding deflection established according to the pipe size and material and the blowdown force using the criteria delineated in Section 3.6.2.2.

B. Non-energy-absorbing members, structural components, and their attachments to the building structure are designed for 2.0 times the restraint reaction to ensure that the required deflection occurs in the energy absorbing members and that the connecting members remain elastic. J E I

All essential components are evaluated for jet impingement and pipe whip effects using a dynamic or an equivalent static analysis of testing to demonstrate either the functional capability and/or operability in addition to the structural integrity of the component. E

3.6.2.3.2.6 Materials

The materials used are as follows:

A. For energy-absorbing members: ASTM A-193 Grade B7 or equivalent for tension rods, and crushable honeycomb made of stainless steel for compression. J

B. For other components: ASTM A-588, ASTM A-572 Grade 50, and ASTM A-36. Charpy tests will be performed on steels subjected to impact loads and lamination tests are performed on steels subjected to through thickness tension.

3.6.2.3.2.7 Jet Impingement Shields E

Protection from jets is provided by using separation and redundancy, as described in Section 3.6.1, in lieu of jet shields.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes to limit pressurization effects in the containment penetration area will not be used except in "Hot Penetration" assemblies as described in Section 3.8.2.1.3.4. J

(Add Section 3.6.2.5)

3.6.2.5 Compartment Pressurization and Temperature Analysis Outside Containment

Energy releases into compartments following a postulated pipe rupture in high and moderate energy lines outside Containment can create pressure differentials across structural walls and slabs and result in adverse environmental conditions for electrical and mechanical equipment located in the compartments. Whereas compartment pressurization analysis is performed to determine pressure loadings on building structures, environmental pressure and temperature analysis is performed to define conditions for equipment qualification. The same basic analytical methods and computer codes are used in both cases, with changes in assumptions and models where appropriate to assure conservative results. Long-term mass and energy releases are used to determine environmental conditions for design and evaluation of equipment and building structures.

The greatest loads on building structures occur shortly after the pipe rupture. The structures are designed to maintain their integrity if such loads were to be imposed on them. The following paragraphs describe the break postulation criteria and calculational techniques used for compartment pressurization and environmental analysis outside Containment.

3.6.2.5.1 Break Postulation Criteria

Break Postulation Criteria for high energy piping is presented in Section 3.6.2.1. For compartment analysis a minimum of one break in each compartment is postulated, and breaks are postulated so as to maximize the adverse effects from pressurization and temperature. When necessary to assure worst conditions, the accident (e.g., the Main Steam System pipe break) is analyzed for a spectrum of pipe break sizes and various plant power conditions.

3.6.2.5.2 Determination of Mass and Energy Release Rates

Piping system energy release transients for the postulated pipe rupture are determined by either a hand calculation or by computer analysis. The plant operating mode which results in the greatest energy release rate is used.

For hand calculations the break mass flow rate is obtained from a critical flow correlation which predicts an upper bound flow rate for the rupture geometry and fluid state under consideration. Examples are the Moody correlation (two-phase and saturated steam conditions), the Homogeneous Equilibrium Model (single phase steam), and the Henry-Fauske correlation (subcooled liquid). Blowdown flow rate is obtained from the following equation per ANSI/ANS-56.10:

$W = C_D A G_C$ where: W = mass flow rate
C_D = discharge coefficient
A = break area
G_C = critical mass flux

The break fluid enthalpy is set equal to the stagnation enthalpy of the fluid in the ruptured pipe. A flow discharge coefficient of 1.0 is used unless a lower value is justified as required by ANSI/ANS-56.10.

For complex systems and where less conservative release rates are needed, computer analysis is employed. Initial conditions (e.g., fluid pressure, fluid temperature) are chosen within normal operating limits such that the set which will result in the largest release rates are used. A system model of appropriate complexity is generated and computer programs of the RELAP4 type are used. To calculate the pipe break response, the fluid system is divided into discrete volumes (control volumes or nodes) which are connected to other volumes by a junction. The equations of conservation of mass and energy are solved in the nodes, and the one-dimensional momentum equation is solved in the flow paths. A time history of system conditions is output by the code. CEFLASH-4A (Ref. Sec. 3.9.1.2.1.22), RELAP4/MOD5, and RELAP5/MOD3 are codes applicable to the generation of mass and energy releases. Also, SGNIII (Ref. Sec. 6.2.1.4.4) may be used in the case of main steam line breaks.

3.6.2.5.3 Compartment Pressurization Analysis and Environmental Pressure and Temperature Analysis

Compartment pressurization analysis is performed to determine pressure loadings on building structures. Environmental pressure and temperature response analysis defines pressure and temperature conditions for qualification of mechanical and electrical equipment.

Computer codes are generally used in some phase of this analysis. Typically the model includes a network of volumes and junctions. Volumes represent rooms, corridors, pipe chases, and other portions of buildings outside Containment. When appropriate, volumes also are used to simulate the HVAC system and outside atmosphere. Junctions represent flow paths between the volumes. multinode analysis may be required within a compartment. The DDIFF-1 addressed below provides acceptable results for both compartment pressurization and environmental pressure and temperature analyses, with appropriate assumptions and models changed to obtain conservative results.

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REFERENCES FOR SECTION 3.6

1. "Evaluation of Potential for Pipe Breaks," NUREG-1061, Vol. 3.

2. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 2 or 3.

3. ASME Code for Pressure Piping, B31, Power Piping, ANSI/ASME B31.1.

4. USNRC Branch Technical Position MEB 3-1 Rev. 2 - Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment, attached to Standard Review Plan 3.6.2, June, 1987.

5. American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture, ANSI/ANS 58.2-1988.

6. R. T. Lahey, Jr. and F. J. Moody, "Pipe Thrust and Jet Loads," The Thermal Hydraulics of a Boiling Water Nuclear Reactor, Section 9.2.3, pp. 375-409, Published by American Nuclear Society, Prepared by the Division of Technical Information United States Energy Research and Development Administration, 1977.

7. RELAP 4/MOD 5, Computer Program User's Manual 098. 026-5.5.

8. USNRC Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems."

9. NUREG/CR-1319, "Cold Leg Integrity Evaluation," Battelle Columbus Laboratories.

10. PICEP: Pipe Crack Evaluation Program, EPRI NP 3596-SR, August, 1984.

11. NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982.

12. "Analysis of Cracked Pipe Weldments," EPRI NP-5057, February 1987.

13. USNRC Regulatory Guide 1.11 (Safety Guide 11), Instrument Lines Penetrating Primary Reactor Containment; including supplement, Backfitting Considerations.

(additional refs.)
INSERT A

Additional References

14. ANSI/ANS-56.10-1982, "Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors," 1982

15. "DDIFF-1 Code, A Description of the DDIFF-1 Digital Computer Code for Reactor Plant Subcompartment Analysis, "Combustion Engineering, Inc., February, 1976.

16. CONTEMPT4/MOD3, "A Multicompartment Containment Systems Analysis Program," (NUREG/CR-2558) E55-2159 EG&G Idaho, Inc., December 1982

Add the following to Secs 3.8.2.5 and 3.8.4.5:

A structural analysis report/s will be prepared for Seismic Category I structures. This report/s will document that the structure's meet the requirements specified in Sec. 3.8 and design changes and identified construction deviations, which could potentially affect the structural capability of the structure, have been incorporated into the structural analysis.

The following records will be reviewed, as applicable:

1) Construction records stating material properties for concrete, reinforcing steel, and structural steel

2) As-built structure dimensions and arrangements

3) Design documents for the structure

Deviations from the design are acceptable provided the following acceptance criteria are met:

1) An evaluation is performed (depending on the extent of the deviations, the evaluation may range from the documenting of an engineering judgement to performance of a revised analysis and design) and

2) The structural design meets the requirements specified in Section 3.8 and

3) The seismic floor response spectra of the as-built structure does not exceed the design basis floor response spectra by more than 10 percent.

The structural analysis report will summarize the results of the reviews, evaluations, and corrective actions, as applicable, and conclude that the as-built structure is in accordance with the design.

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3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 CONCRETE CONTAINMENT

This section is not applicable to the System 80+ Standard Design. For a description of the containment, see Section 3.8.2. For a description of the containment shield building, see Section 3.8.4.

3.8.2 STEEL CONTAINMENT

3.8.2.1 Description of the Containment

3.8.2.1.1 General

The containment is a spherical free-standing welded steel structure. The sphere is supported by sandwiching its lower portion between the building foundation concrete and the interior structure base. There is no structural connection either between the containment and the interior structure, or between the containment and the shield building. The diameter of containment is 200 ft. The plate nominal thickness is 1.75 inches. The anchorage region plate thickness is 2 inches. The containment is shown on the plans and elevations of Figures 1.2-2, 1.2-3, 1.2-5, 1.2-6, 1.2-7 and 1.2-9.

The spherical shell plate segments will be shop fabricated and field welded. These plates will be approximately 25 feet long and 15 feet wide and can weigh as much as ten tons each; however, these dimensions will vary depending upon the plate location. Two or more plates may be assembled and field welded on the ground and then erected. A vast majority of penetration assemblies will be shop welded to the vessel plates, while others will be attached to the vessel in the field. Vessel plate will be thickened around the penetration to compensate for the openings. Where there is a cluster of penetrations in the same plate segment, the entire segment may be fabricated out of the thicker plate, tapered to 1.75 inches at the edges. The additional thickness will depend upon the nominal size, thickness and location of the penetration sleeve and shall be in accordance with ASME Boiler and Pressure Vessel Code (ASME Code) requirements (Reference 1).

The 2 inch thick portion of the steel containment vessel in the anchorage region will be shop fabricated and welded. The longitude plate welds will be 2 inch welds and will be postweld heat treated. The top and bottom edges of these 2 inch plates will be tapered to 1.75 inches.

The ~~basic configuration~~ arrangement of the Nuclear Island structures, which includes containment and defines critical dimensions, flood barriers, and fire barriers, is shown on Figure 3.8-5, Shs. 1-12.

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3.8.2.1.2 Anchorage Region

The containment is assumed to behave as an independent, free-standing structure above elevation 91+9. Below elevation 91+9, the vessel is encased between the base slab of the internal structures and the shield building foundation. In the transition region, a compressible material is provided as shown in Figure 3.8-1 to eliminate excessive bearing loads on the concrete as well as to reduce the secondary stresses in the vessel at this location. No shear connectors are provided between the containment plate and the shield building foundation or base slab of internal structures. The lateral loads due to seismic forces, etc., are transferred to the foundation concrete by friction and bearing. The containment shell is thickened to 2 inches in the anchorage region for corrosion allowance. The vessel plate thickness in the embedded zone is the same as in the free zone.

3.8.2.1.3 Containment Penetrations

3.8.2.1.3.1 Equipment Hatch

The equipment hatch is composed of a cylindrical sleeve in the containment shell and a dished head 22 feet in diameter with mating bolted flanges. The flanged joint has double seals with an annular space for pressurized leak testing in accordance with 10CFR50, Appendix J.

The equipment hatch is designed, and fabricated in accordance with Section III, Subsection NE of the ASME Boiler and Pressure Vessel Code. The equipment hatch is tested and stamped with the containment vessel.

Details of a typical equipment hatch are shown on Figure 3.8-1.

3.8.2.1.3.2 Personnel Locks

Two personnel locks 10 feet in diameter are provided for each unit. Each lock has double doors with an interlocking system to prevent both doors being opened simultaneously. Remote indication is provided to indicate the position of each door. Double seals are provided on each door with an annular space for pressurized leak testing in accordance with 10CFR50, Appendix J.

The personnel locks are welded steel subassemblies designed, fabricated, tested, and stamped in accordance with Section III, Subsection NE of the ASME Code.

Details of a typical personnel lock are shown on Figure 3.8-1.

Seals are designed to maintain containment integrity for Design Basis Accident conditions, including pressure, temperature, and radiation.

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All electrical penetrations, including seals, are designed to maintain containment integrity for Design Basis Accident conditions, including pressure, temperature, and radiation. Double barriers permit testing of each assembly in accordance with 10CFR50 Appendix J to verify that containment integrity is maintained.

The electrical penetration assemblies are designed, fabricated, tested, and stamped in accordance with IEEE-317. The pressure boundary portion of the assembly is designed, fabricated, tested and stamped in accordance with Section III, Subsection NE of the ASME Code.

3.8.2.2 Applicable Codes, Standards, and Specifications

The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of the steel containment and penetrations is covered by the following codes, standards, specifications, and regulations:

Codes	Title
ASME	Boiler and Pressure Vessel Code, Section II, "Material Specifications"
ASME	Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components"
ASME	Boiler and Pressure Vessel Code, Section V, "Nondestructive Examination"
ASME	Boiler and Pressure Vessel Code, Section IX, "Welding and Brazing Qualifications"
ASME	Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Subsection IWE "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants"

With either option care must be taken to prevent the floating of the containment. This is accomplished by filling the containment with water as the grout/concrete is placed. After the lower containment section has been placed, the construction of the interior structure can begin. The assembly of the containment sections will continue in the yard. As the work on the interior structure continues additional sections of the containment can be lifted into place. After the major equipment is placed in the interior structure the top section of the containment vessel can be set. The completed containment will then be used to support the scaffolding for the concrete dome of the shield building.

3.8.2.7 Testing and In-service Surveillance Requirements

The containment vessel, personnel airlocks and equipment hatch are inspected and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection NE. Penetrations are pressure tested as required for Subsection NC of the ASME Code.

Periodic leakage rate tests of the containment are conducted in accordance with 10CFR50, Appendix J to verify leak tightness and integrity. These tests and other in-service inspection requirements are described in Section 6.2. Periodic in-service inspections are conducted in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE.

3.8.3 CONCRETE AND STRUCTURAL STEEL INTERNAL STRUCTURES

3.8.3.1 Description of the Internal Structures

The internal structure is a group of reinforced concrete structures that enclose the reactor vessel and primary system. The internal structure provides biological shielding for the containment interior. The internal structure concrete base rests inside the lower portion of the containment vessel sphere. A description of various structures that constitute the internal structure is given in the following paragraphs. The details of the internal structure are shown in Figures 1.2-2, 1.2-3, 1.2-6, 1.2-7 and 1.2-9.

The primary shield wall encloses the reactor vessel and provides protection for the vessel from internal missiles. The primary shield wall provides biological shielding and is designed to withstand the temperatures and pressures following LOCA. In addition, the primary shield wall provides structural support for the reactor vessel. The primary shield wall is a minimum of six feet thick.

The arrangement ~~basic configuration~~ of the Nuclear Island Structures, which includes the internal structure and defines critical dimensions, flood barriers, and fire barriers, is shown on Figure 3.8-5, Shs. 1-12.

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The secondary shield wall (crane wall) provides supports for the polar crane and protects the steel containment vessel from internal missiles. In addition to providing biological shielding for the coolant loop and equipment, the crane wall also provides structural support for pipe supports/restraints and platforms at various levels. The crane wall is a right cylinder with an inside diameter of 130 feet and a height of 118 feet from its base. The crane wall is a minimum of four feet thick.

The refueling cavity, when filled with borated water, facilitates the fuel handling operation without exceeding the acceptable level of radiation inside the containment. The refueling cavity has the following sub-compartments:

A. Storage area for upper guide structure.

B. Storage area for core support barrel.

C. Refueling canal.

The refueling canal, when filled with borated water, forms a pool above the reactor vessel. The reactor vessel flange is permanently sealed to the bottom of the refueling canal to prevent leakage of refueling water into the reactor cavity. The fuel transfer tube connects the refueling canal to the Spent Fuel Pool. The refueling canal is filled with borated water to a depth that limits the radiation at the surface of the water to acceptable levels during the period when a fuel assembly is being transferred to the Spent Fuel Pool. The shield walls that form the refueling cavity are a minimum of six feet thick.

The In-containment Refueling Water Storage Tank (IRWST) provides storage of refueling water, a single source of water for the safety injection and containment spray pumps and a heat sink for the Safety Depressurization System. The IRWST is dishlike in shape and utilizes the lower section of the Internal Structure as its outer boundary. The IRWST is provided with a stainless steel liner to prevent leakage. A full description of the IRWST is provided in Section 6.3.2.

The operating floor provides access for operating personnel functions and provides biological shielding. Inside the crane wall, the operating floor is a reinforced concrete slab with a covered hatch that is aligned with hatches in the two lower floors. Outside the crane wall, the operating floor consists of steel grating. There are also reinforced concrete floor slabs at elevation 115+6 and elevation 91+9 that connect the crane wall and the primary shield wall.

Design of the IRWST ~~will~~ considers pressurization as a result of the containment systems Design Basis Accident.

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3.8.3.4 <u>Design and Analysis Procedures</u>

The internal structure is designed for the loads and load combinations specified in Section 3.8.3.3. The complete internal structure (and supporting substructure) is modeled with three-dimensional solid, plate or shell and beam finite elements using ANSYS or another suitable computer code. The forces and moments resulting from the applied static and dynamic loads are used to design the walls, slabs, beams and columns which make up the Internal Structure. The design is performed using either ACI 349 (Reference 3) or ANSI/AISC N690 (Reference 4) as appropriate.

3.8.3.5 <u>Structural Acceptance Criteria</u>

The structural acceptance criteria for the Internal Structures is outlined in Section 3.8.4.5.

3.8.3.6 <u>Materials, Quality Control, and Special Construction Techniques</u>

Insert 1 ~~The~~ Additional materials, quality control, and special construction techniques for the concrete internal structures are outlined in Section 3.8.4.6.

3.8.3.7 <u>Testing and In-service Surveillance Requirements</u>

Testing and in-service surveillance requirements are outlined in Section 3.8.4.7.

3.8.4 OTHER CATEGORY I STRUCTURES

3.8.4.1 <u>Description of the Structures</u>

3.8.4.1.1 Reactor Building

The reactor building is composed of the containment shield building, steel containment vessel including the internal structures, and subsphere. The steel containment vessel is described in Section 3.8.2. The internal structures are described in Section 3.8.3. Details of the reactor building are shown in Figures 1.2-2 through 1.2-10.

Insert 2 The containment shield building is a reinforced concrete structure composed of a right cylinder with a hemispherical dome. The containment shield building shares a common foundation base mat with the nuclear system annex. The containment shield building houses the steel containment vessel and safety-related equipment located in the subsphere, and is designed to provide biological shielding as well as external missile protection for the steel containment shell and safety-related equipment.

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INSERT 1

The design will address the vertical alignment of the Secondary Shield Wall (Crane Wall) with the corresponding structure below the containment and provide special construction tolerances, as necessary, to ensure potential misalignment is appropriately considered.

INSERT 2

The arrangement ~~basic configuration~~ of the Nuclear Island Structures, which includes the reactor building and defines critical dimensions, flood barriers, and fire barriers, is shown on Figure 3.8-5, Shs. 1-12.

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The containment shield building has an inner radius of 105 feet, a cylinder thickness of 4 feet up to elevation 146+0. Above elevation 146+0 the shield building thickness is 3 feet including the dome area. The height of the containment shield building is approximately 215 feet. The structural outline of the containment shield building is shown on Figures 1.2-2 and 1.2-3. An annular space is provided between the containment vessel and containment shield building above elevation 91+9 for structural separation and access to penetrations for testing and inspection. The shield building and the nuclear annex are connected to form a monolithic structure.

The subsphere is that portion of the reactor building which is below elevation 91+9 and external to the containment vessel. The subsphere houses auxiliary safety-related equipment. This area below the spherical containment allows efficient use of space for locating safety equipment adjacent to the containment vessel and eliminating excessive piping while allowing maximum access to the containment for locating penetrations.

3.8.4.1.2 Nuclear System Annex

The Nuclear System Annex is composed of the control complex, diesel generator areas, main steam valve house areas, CVCS and maintenance areas and spent fuel storage area.

The nuclear system annex is a reinforced concrete structure composed of rectangular walls, columns, beams, and floor slabs. The nuclear system annex shares common walls and foundation basemat with and is monolithically connected to the containment shield building. In addition to the structural components, there are components designed to provide biological shielding and protection against tornado and turbine missiles. Structural components, as well as members serving as shielding components, vary in thickness from approximately one foot to five feet. Details of the nuclear system annex are shown in Figures 1.2-2 through 1.2-10.

3.8.4.1.3 Station Service Water System Structure

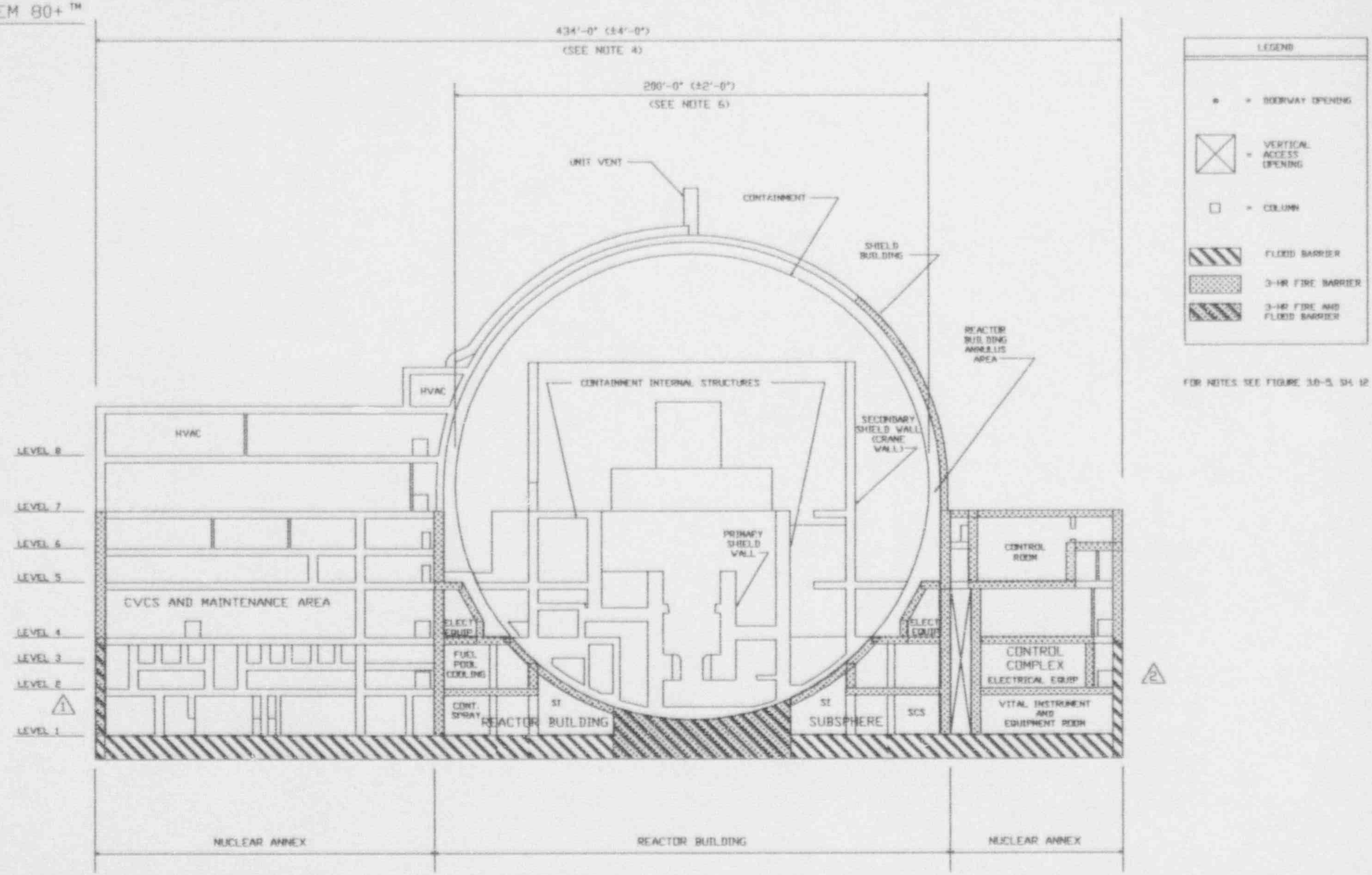
The station service water structure is a concrete structure which is separately located from the Nuclear Island. The location is site specific. The structure contains safety related equipment.

3.8.4.2 Applicable Codes, Standards, and Specifications

Category I structures are designed in accordance with the codes and criteria shown in Table 3.8-4.

The ~~basic configuration~~ arrangement of the Nuclear Island structures, which includes the nuclear system annex and defines critical dimensions, flood barriers, and fire barriers, is shown on Figure 3.8-5, Shs 1-12.

SYSTEM 80+™



FOR NOTES SEE FIGURE 3.8-5, SH. 12

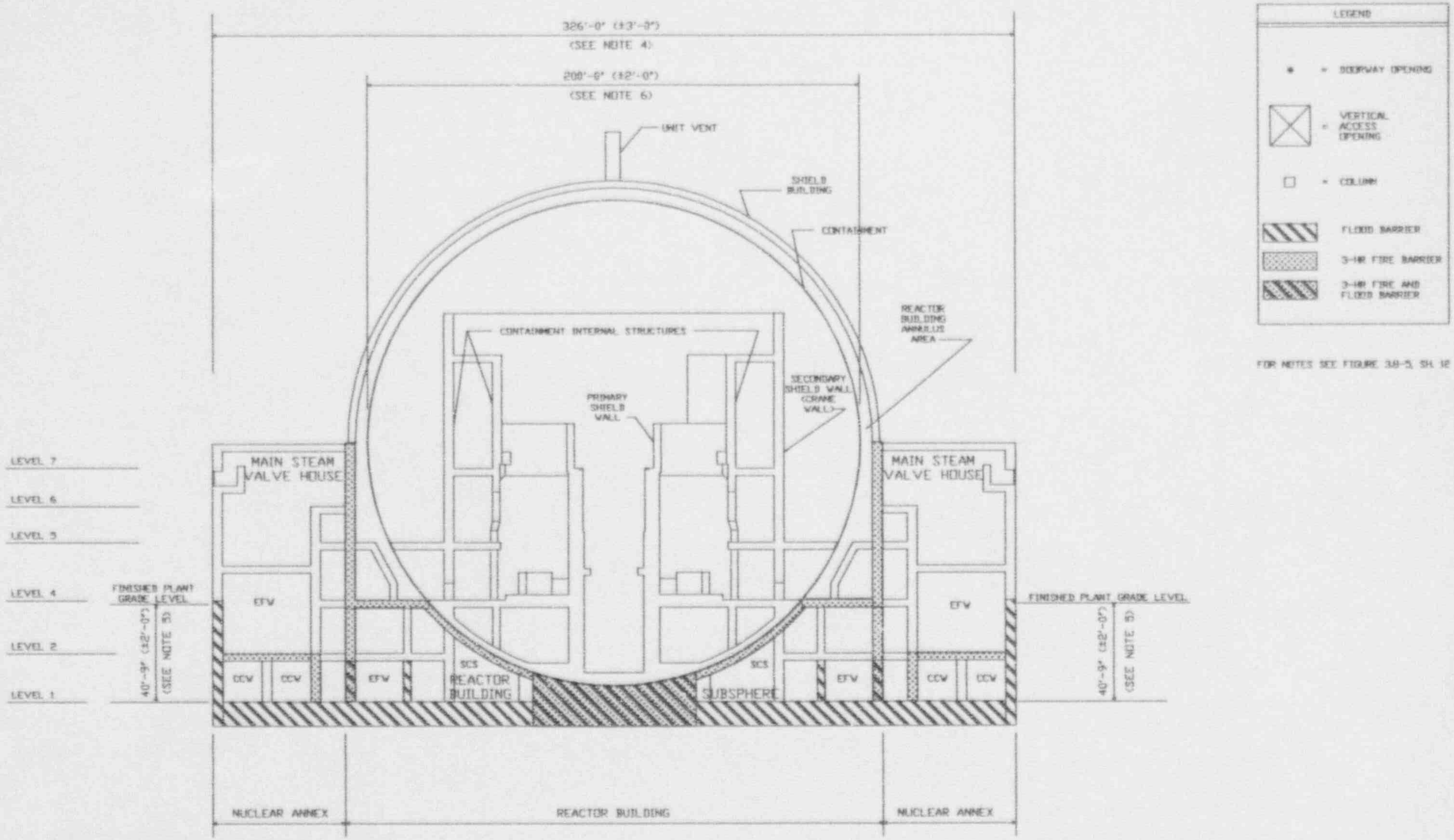
1. THE RADIOACTIVE WASTE STRUCTURE IS LOCATED ADJACENT TO THE NUCLEAR ANNEX

2. THE TURBINE BUILDING IS LOCATED ADJACENT TO THE NUCLEAR ANNEX

NUCLEAR ISLAND STRUCTURES
SECTION A-A

FIGURE 3.8-5
SH. 1 OF 12

SYSTEM 80+ ™

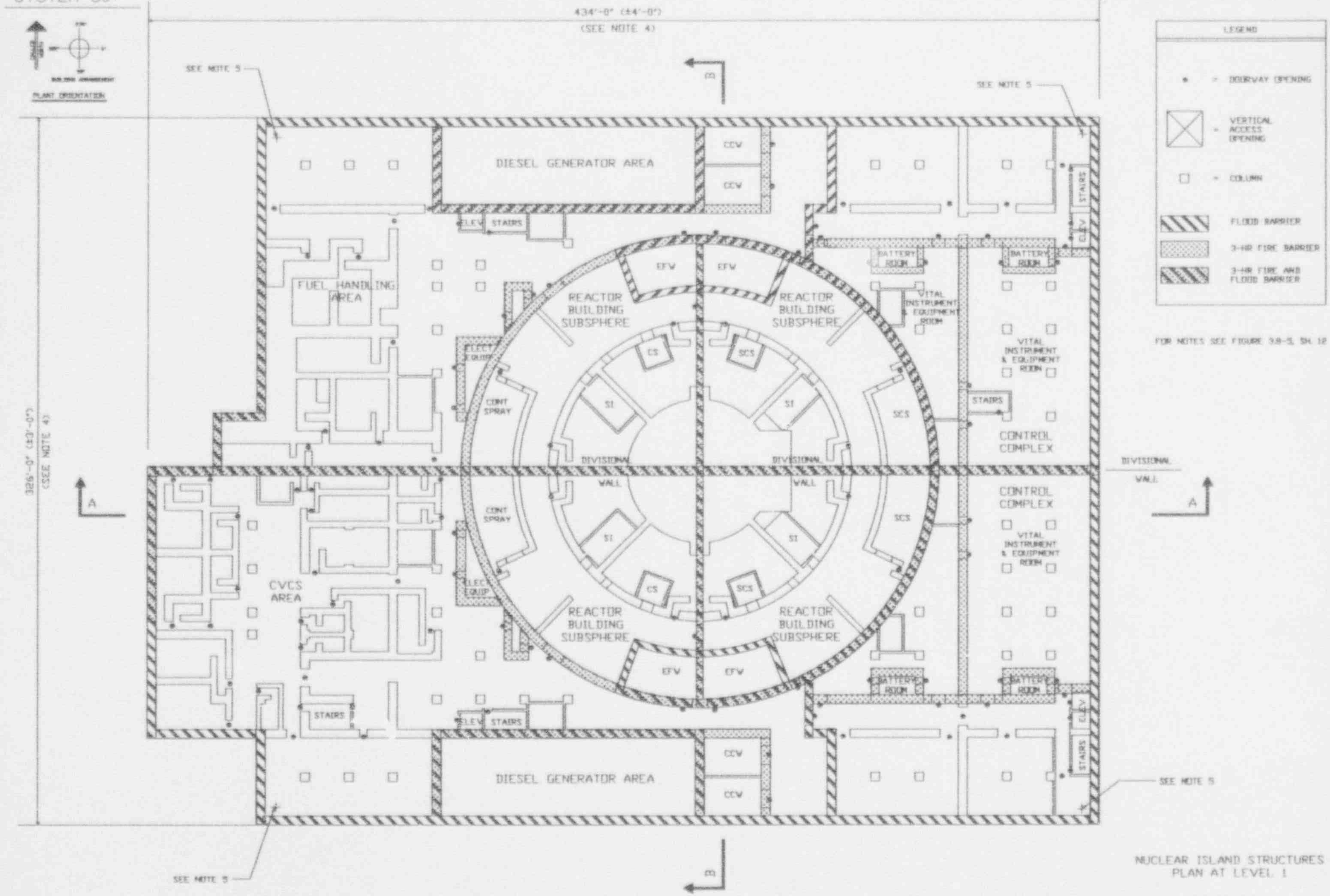


FOR NOTES SEE FIGURE 3.8-5, SH. 12

NUCLEAR ISLAND STRUCTURES
SECTION B-B

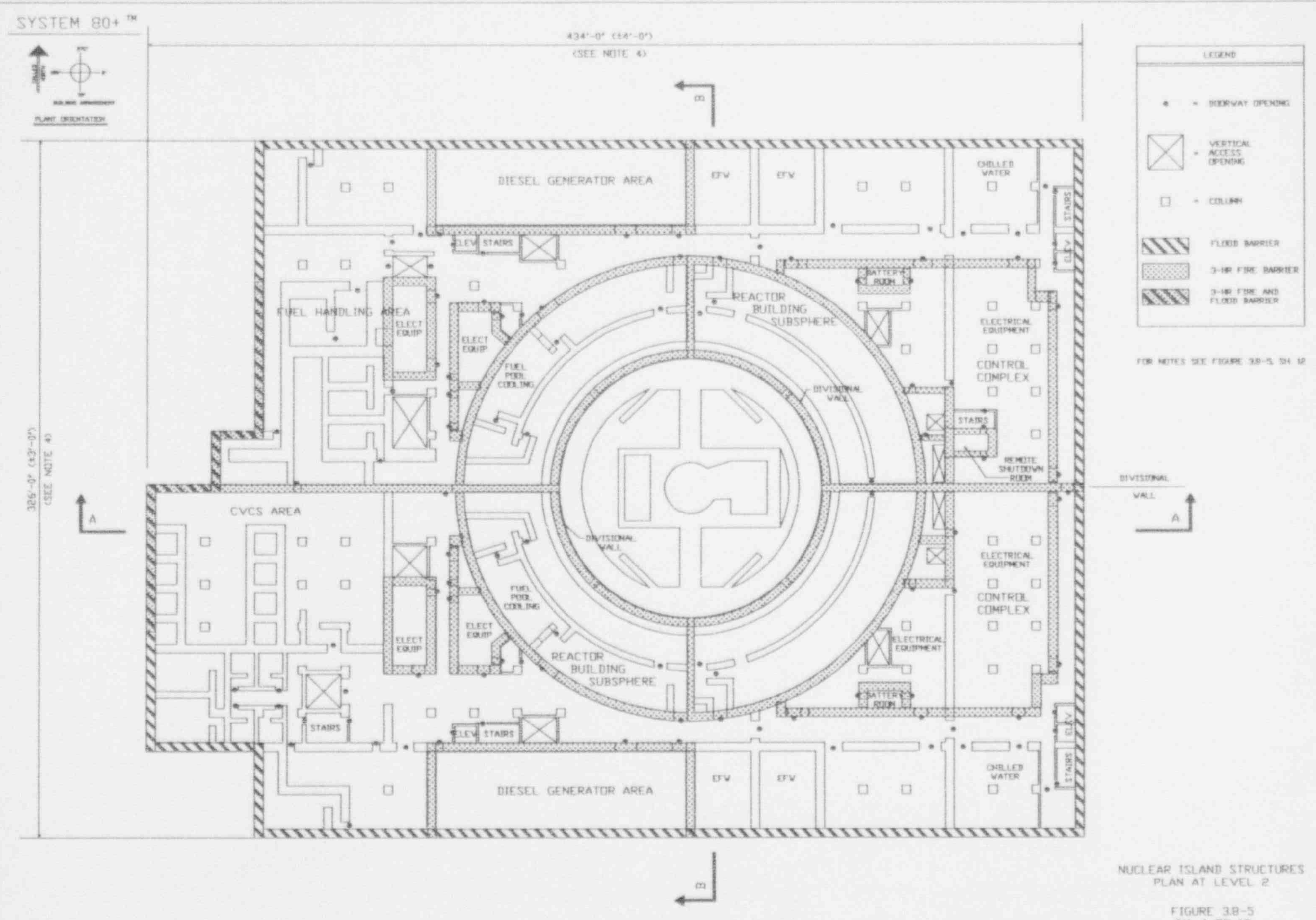
FIGURE 3.8-5
SH. 2 OF 12

SYSTEM 80+™



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 1

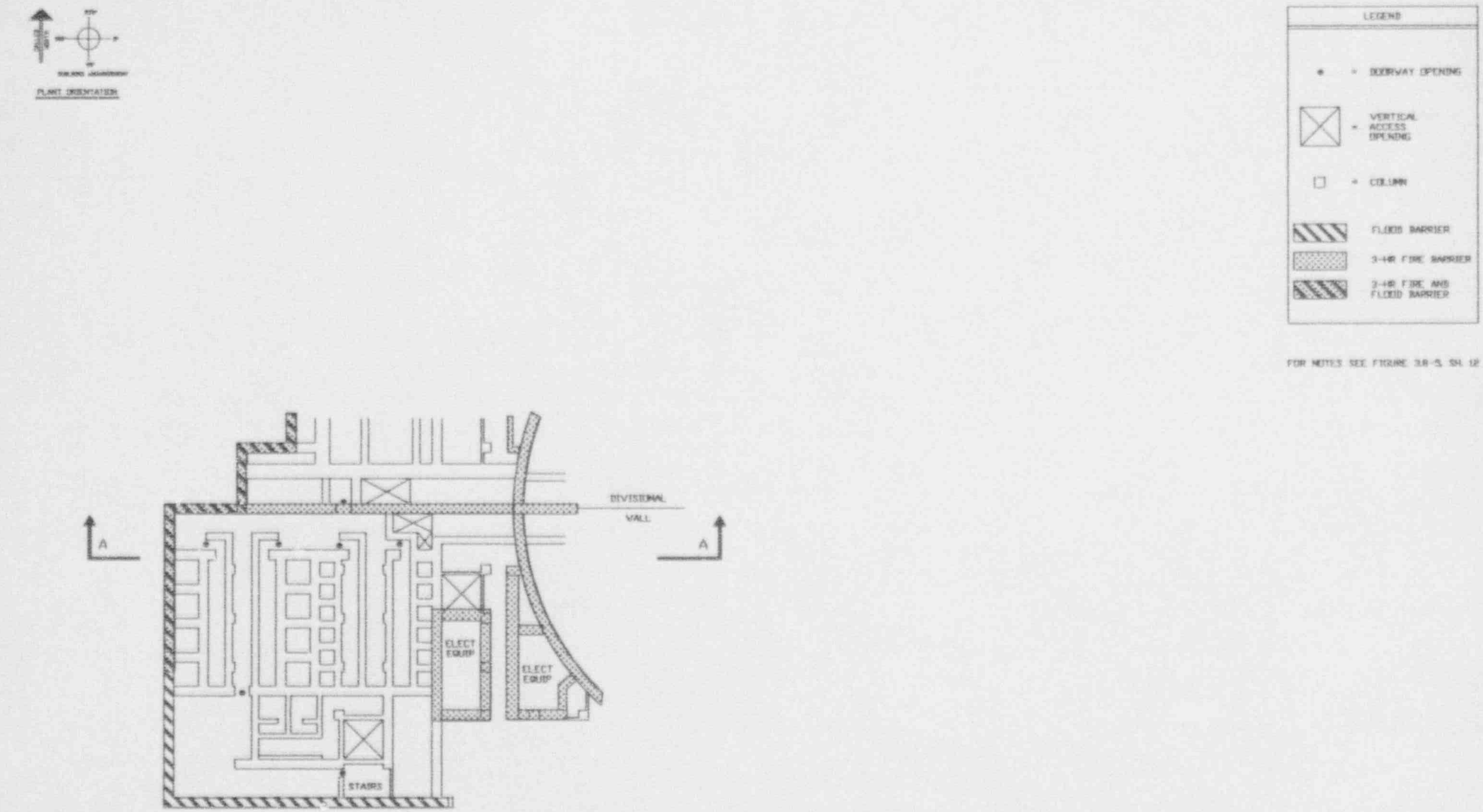
FIGURE 3.8-5
SH. 3 OF 12



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 2

FIGURE 3.8-5
SH. 4 OF 12

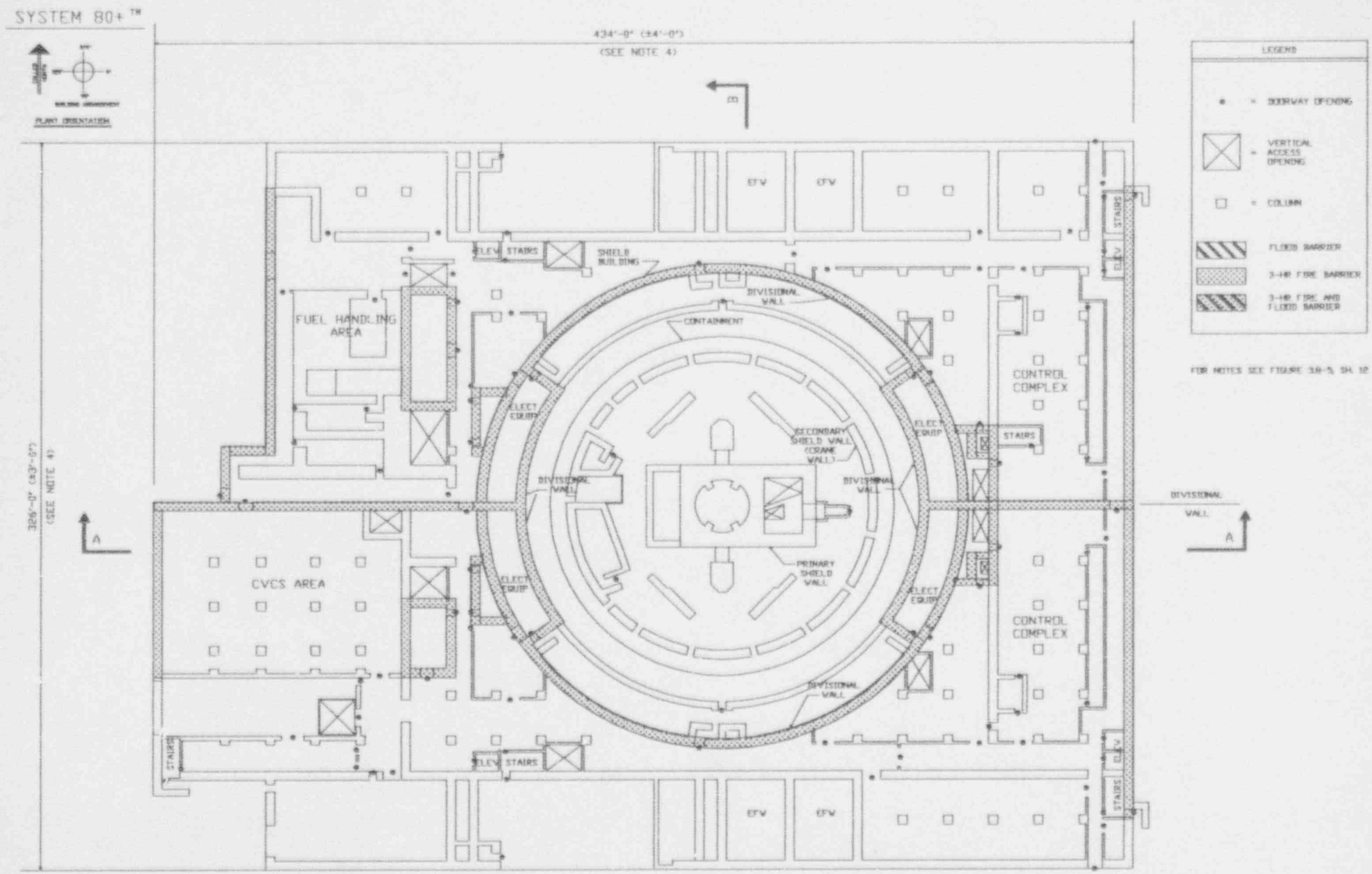
SYSTEM 80+ ™



FOR NOTES SEE FIGURE 3.8-5, SH. 12

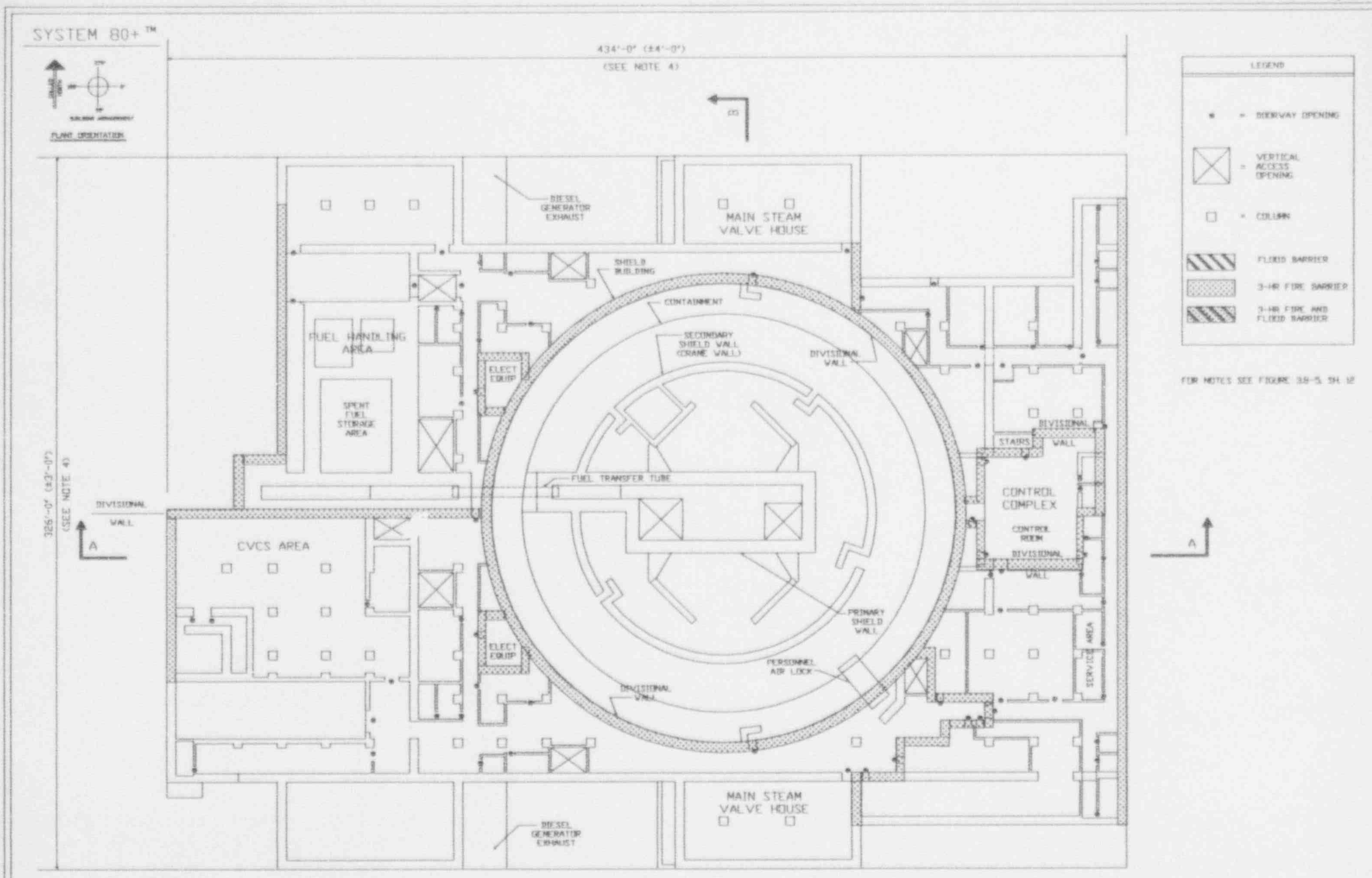
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 3

FIGURE 3.8-5
SH. 5 OF 12



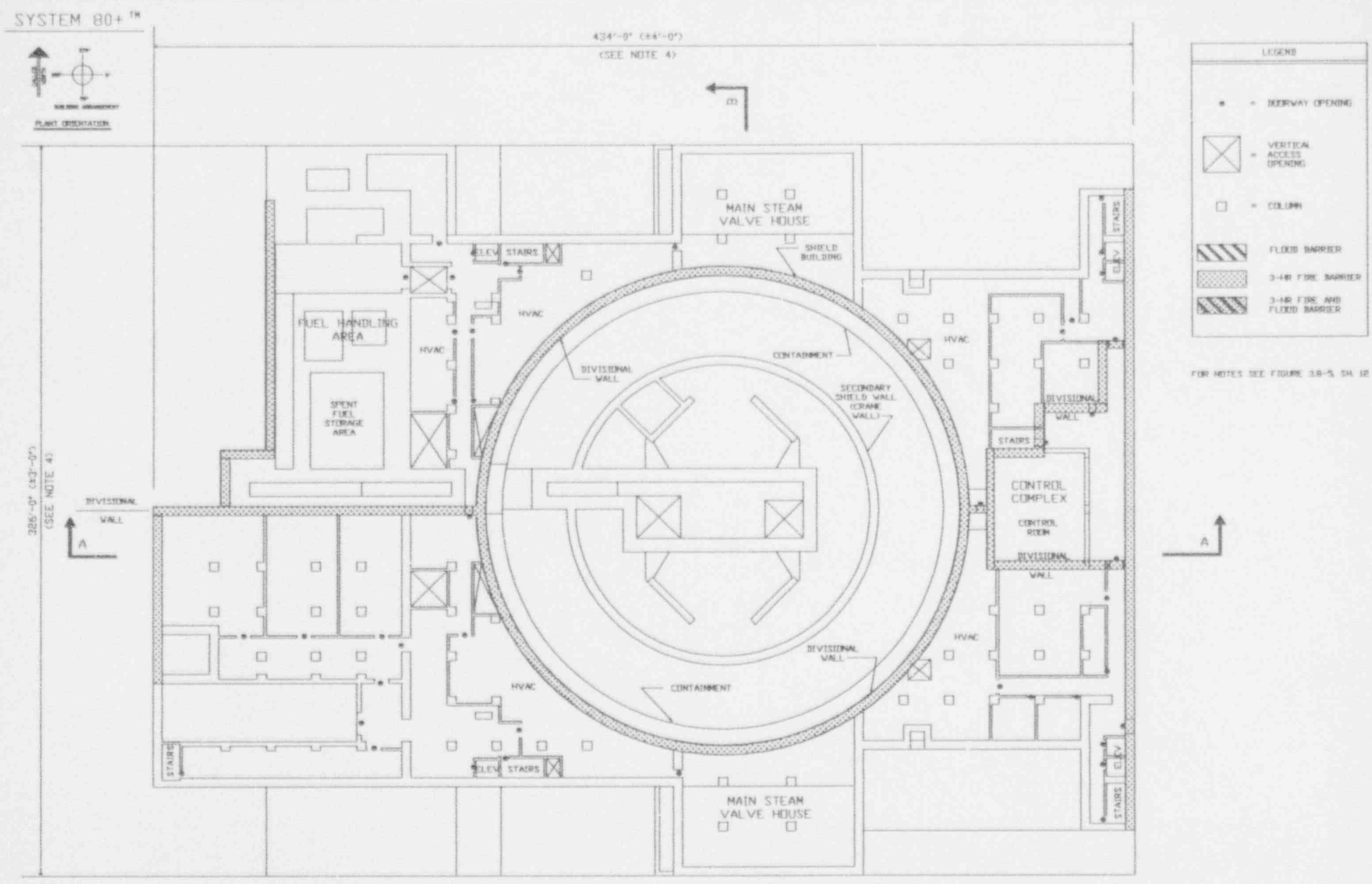
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 4

FIGURE 3.8-5
SH. 6 OF 12



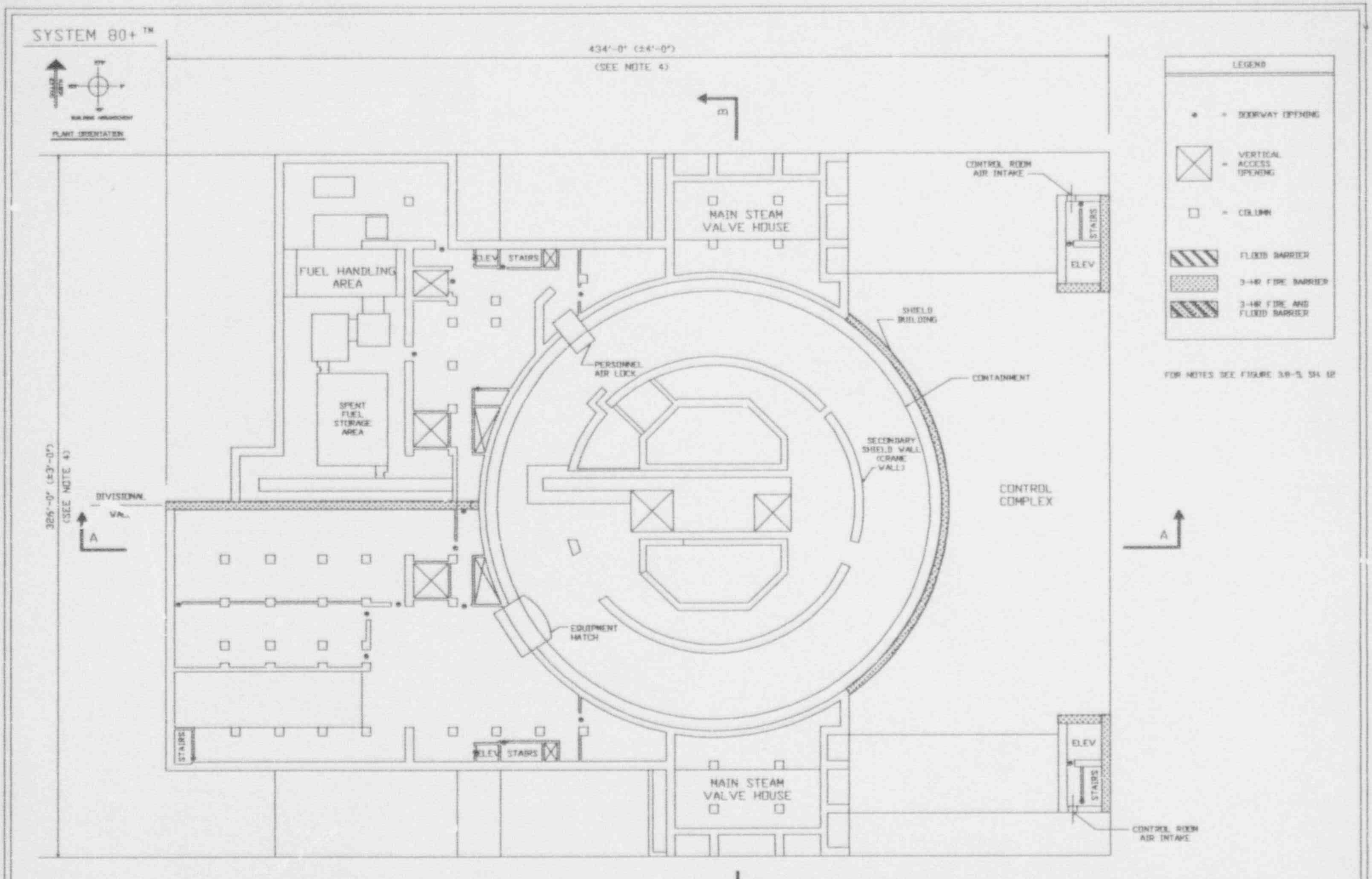
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 5

FIGURE 3.8-5
SH. 7 OF 12



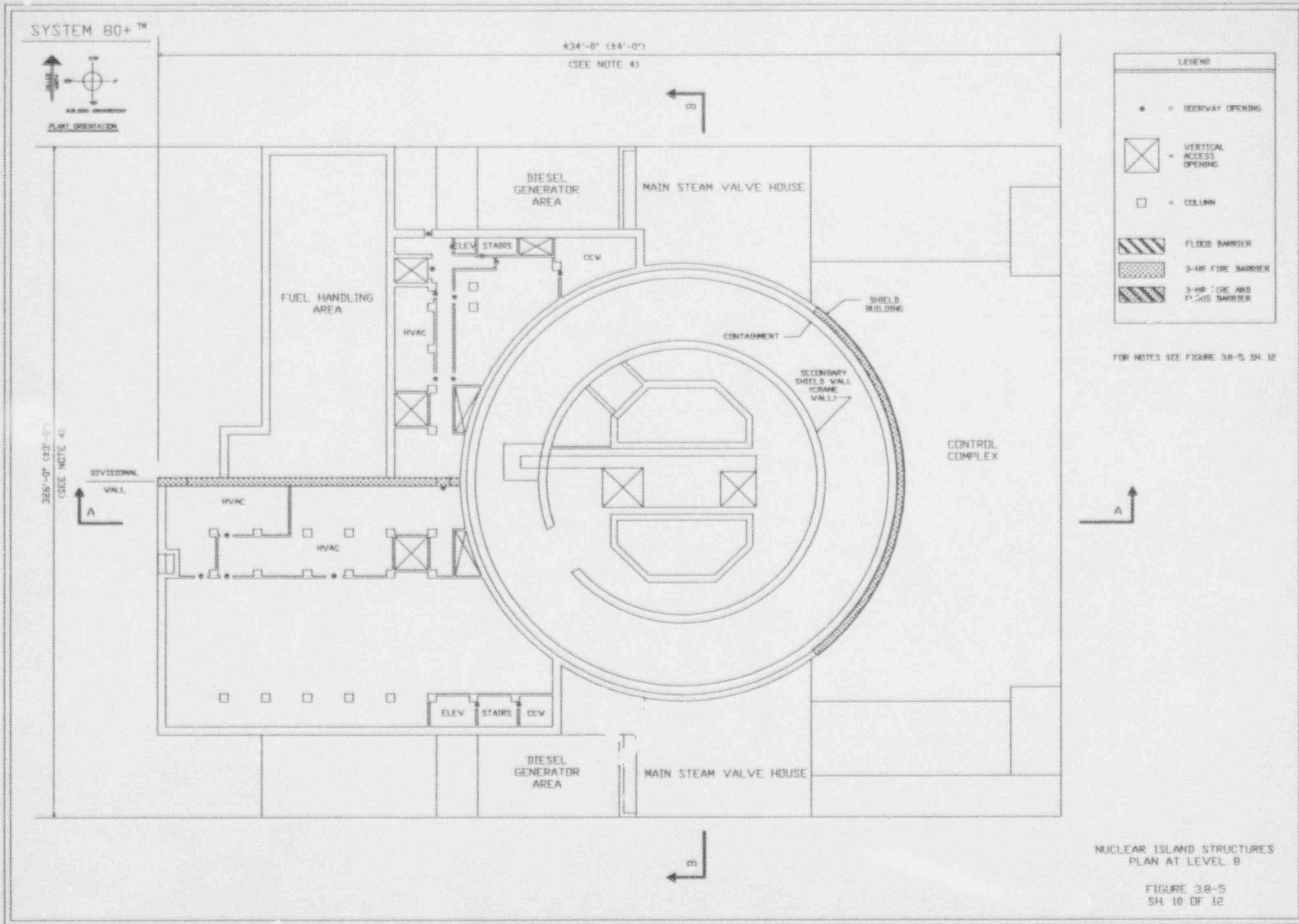
NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 6

FIGURE 3.8-5
SH. 8 OF 12



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL 7

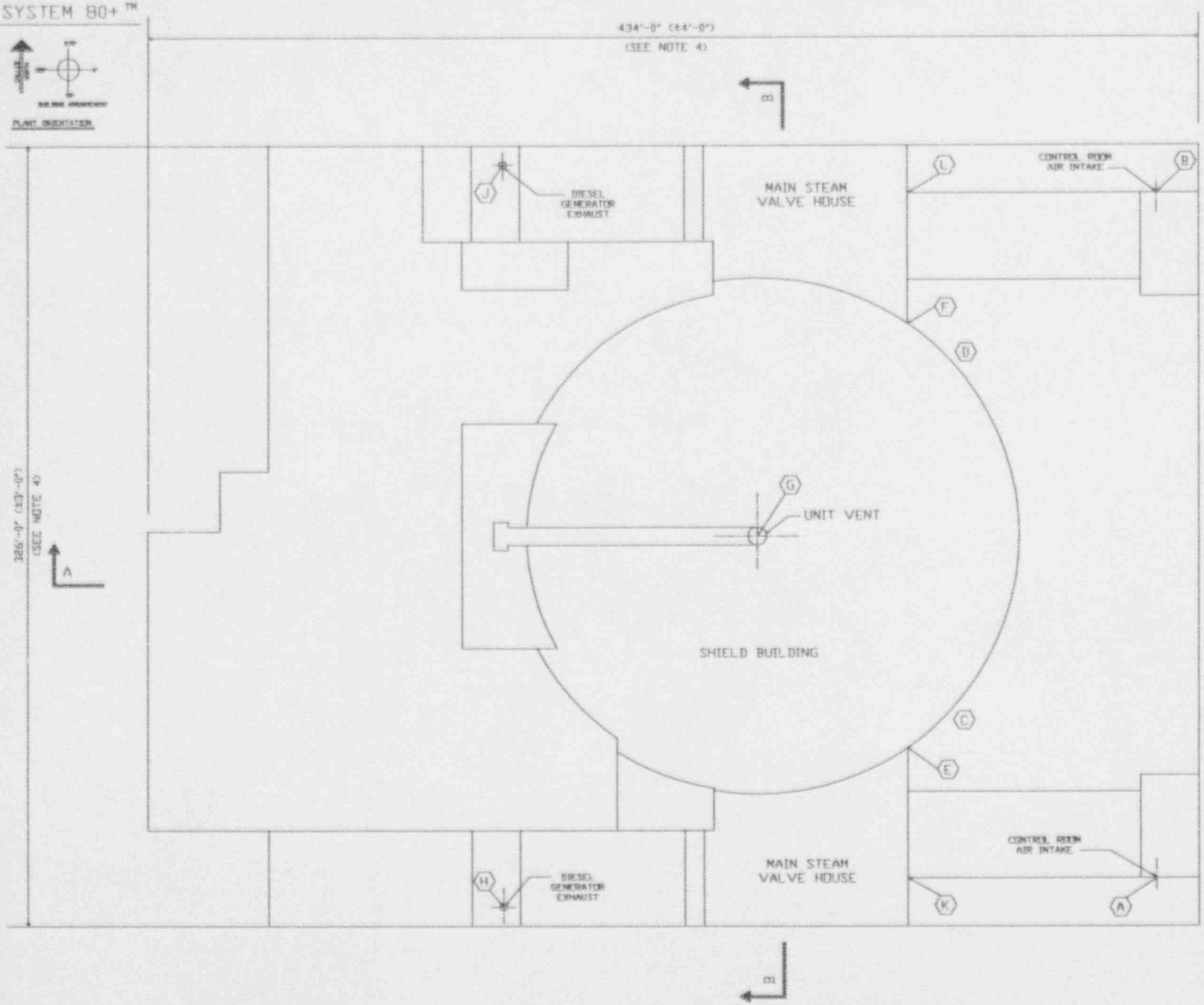
FIGURE 3.8-5
SH. 9 OF 12



NUCLEAR ISLAND STRUCTURES
PLAN AT LEVEL B

FIGURE 3.8-5
SH. 10 OF 12

SYSTEM 80+™



LEGEND

- • = DOORWAY OPENING
- ☒ = VERTICAL ACCESS OPENING
- □ = COLUMN
- FLOOD BARRIER
- 3-HR FIRE BARRIER
- 3-HR FIRE AND FLOOD BARRIER

FOR NOTES SEE FIGURE 3.8-5, SH. 12

MINIMUM DISTANCES FROM CONTROL ROOM AIR INTAKES (SEE NOTE 7)

LOCATION TO LOCATION		MINIMUM DISTANCE
A	C	183'-0"
A	F	250'-0"
A	K	63'-0"
A	G	211'-0"
A	H	150'-0"
A	J	150'-0"
B	E	250'-0"
B	D	183'-0"
B	L	63'-0"
B	G	211'-0"
B	J	156'-0"
B	K	150'-0"

NUCLEAR ISLAND STRUCTURES
PLAN AT ROOF

FIGURE 3.8-5
SH. 11 OF 12

SYSTEM 80+™

NOTES FOR FIGURES:

1. FLOOD DOORS ARE PROVIDED IN FLOOD BARRIERS, AND PENETRATIONS ARE SEALED UP TO THE EXTERNAL AND INTERNAL FLOOD LEVELS. SENSORS ARE PROVIDED ON FLOOD DOORS WITH OPEN AND CLOSE STATUS INDICATIONS AT A MONITORED LOCATION.

2. 3-HOUR FIRE RATED DOORS AND ELECTRICAL AND MECHANICAL PENETRATION SEALS ARE PROVIDED FOR OPENINGS IN THE 3-HOUR FIRE RATED BARRIERS.

3. THE FOLLOWING STRUCTURES, SYSTEMS, AND COMPONENTS DEPICTED ON THESE FIGURES ARE NOT SEISMIC CATEGORY I:
   - DOORWAY OPENINGS
   - VERTICAL ACCESS OPENINGS
   - STAIRS
   - ELEVATORS

4. THIS DIMENSION IS MEASURED AT THE TOP ELEVATION OF THE LEVEL 4 REINFORCED CONCRETE FLOOR (±0'-6") IN A DIRECTION PARALLEL TO THE RESPECTIVE PLANT ORIENTATION AZIMUTH, 0° - 180° (±2°) OR 90° - 270° (±2°), BETWEEN THE EXTERIOR SURFACES OF THE REINFORCED CONCRETE AT THE CORNERS SHOWN.

5. THIS DIMENSION IS THE DIFFERENCE BETWEEN THE PLANT GRADE ELEVATION AND THE TOP ELEVATION OF THE LEVEL 1 REINFORCED CONCRETE FLOOR AT THE LOCATIONS INDICATED ON FIGURE 3.8-5, SH. 3. THE PLANT GRADE ELEVATION IS DETERMINED AT THE EXTERIOR CORNER OF THE REINFORCED CONCRETE WALL ADJACENT TO THE LOCATIONS INDICATED ON FIGURE 3.8-5, SH. 3.

6. 200'-0" (±2'-0") IS THE INSIDE DIAMETER OF THE STEEL CONTAINMENT SPHERE. THE INSIDE RADIUS OF THE SPHERE IS 100'-0" (±1'-0"). THE INSIDE RADIUS IS MEASURED AT THE ELEVATION OF THE CENTER OF THE SPHERE (±0'-6") IN FOUR DIRECTIONS, PLANT ORIENTATION AZIMUTHS 0° (±5°), 90° (±5°), 180° (±5°), 270° (±5°). ONE ADDITIONAL INSIDE RADIUS IS MEASURED FROM THE CENTER OF THE SPHERE, VERTICALLY (±5°), TO THE TOP OF THE CONTAINMENT.

7. THE MINIMUM DISTANCE IS MEASURED IN A HORIZONTAL DIRECTION AT THE ELEVATION OF THE CENTERLINE OF THE CONTROL ROOM AIR INTAKE (±0'-6") BETWEEN THE IDENTIFIED LOCATIONS. THE LOCATIONS ARE FURTHER DESCRIBED AS FOLLOWS:

   A AND B - THE INTERSECTION OF THE CENTERLINE OF THE CONTROL ROOM AIR INTAKE WITH THE PLANE OF THE EXTERIOR REINFORCED CONCRETE NUCLEAR ANNEX WALL.

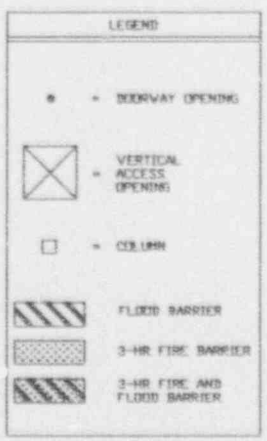
   C AND D - THE POINT ALONG THE LINE FORMED BY THE INTERSECTION OF THE REINFORCED CONCRETE SHIELD BUILDING EXTERIOR WALL AND THE EXTERIOR SURFACE OF THE REINFORCED CONCRETE NUCLEAR ANNEX ROOF THAT IS CLOSEST TO THE CONTROL ROOM AIR INTAKE.

   E AND F - THE INTERSECTION OF THE REINFORCED CONCRETE MAIN STEAM VALVE HOUSE EXTERIOR WALL, THE REINFORCED CONCRETE SHIELD BUILDING EXTERIOR WALL, AND THE EXTERIOR SURFACE OF THE REINFORCED CONCRETE NUCLEAR ANNEX ROOF.

   G - THE INTERSECTION OF THE CENTERLINE OF THE UNIT VENT WITH THE PLANE PASSING THROUGH THE TOP SURFACE OF THE UNIT VENT.

   H AND J - THE INTERSECTION OF THE CENTERLINE OF THE DIESEL GENERATOR EXHAUST WITH THE PLANE PASSING THROUGH THE TOP SURFACE OF THE DIESEL GENERATOR EXHAUST.

   K AND L - THE POINT ALONG THE LINE FORMED BY THE INTERSECTION OF THE REINFORCED CONCRETE MAIN STEAM VALVE HOUSE EXTERIOR WALL AND THE EXTERIOR SURFACE OF THE REINFORCED CONCRETE NUCLEAR ANNEX ROOF THAT IS CLOSEST TO THE CONTROL ROOM AIR INTAKE.



ABBREVIATIONS:

BLDG	BUILDING
CONT	CONTAINMENT
ELECT	ELECTRICAL
ELEV	ELEVATOR
EQUIP	EQUIPMENT
HR	HOUR
MAINT	MAINTENANCE
SYS	SYSTEM

NUCLEAR ISLAND STRUCTURES
NOTES, LEGEND,
AND ABBREVIATIONS

FIGURE 3.8-5
SH. 12 OF 12

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ANSYS or another suitable computer code on an elastic foundation. The forces and moments determined in the analysis are input to the structural design using ACI 349.

3.8.5.5 Structural Acceptance Criteria

These are outlined in Section 3.8.4.5.

3.8.5.6 Material, Quality Control, and Special Construction Techniques

These are outlined in Section 3.8.4.6.

3.8.5.7 Testing and In-service Surveillance Requirements

These are outlined in Section 3.8.4.7.

The analysis and design of the foundations ~~will~~ considers the effects of varying soil properties beneath a specific foundation and the effects of construction sequence with particular emphasis on differential settlements of the basemat.

A settlement monitoring program is required ~~will be established~~ for all Seismic Category I structures. Settlement monuments will be provided at appropriate locations to track total and differential settlements. Monitoring will begin as each monument is installed. Actual vs predicted settlements will be tracked and evaluated for each Seismic Category I structure.

The Operation Support Center, Men's Change, Women's Change, Break Room, Shift Assembly and Offices, Radiation Access Control and Cas. and Sec. Group areas all are served by an individual air handling unit consisting of a centrifugal fan, non-essential chilled water coil and roughing filter. Two non-essential electrical and CEDM control rooms are served by two 100% air handling units consisting of a centrifugal fan, non essential chilled water coil and roughing filter. J

As shown on Figure 9.4-2 all of these areas can receive outside air from the cleanest of two sources described for the control room. I The roof exhaust fan shown serving the men's change, the women's change and the break room is actually located at least 80 feet from the outside air intake. J

9.4.1.3 **Safety Evaluation**

The air-handling system serving the control room proper consists of two completely redundant, independent, full-capacity cooling systems. Each system is powered from independent, Class 1E power sources and headered on separate essential chilled water systems.

Equipment capacities are selected based on conservative evaluations of heat-producing equipment and conservative assumptions of adjacent area temperatures. Normally, the control room temperature will be maintained at approximately 74°F. The design basis upper limit of 85°F is set to insure reliable operation of the electronic equipment. E

Both, the Technical Support Center and computer room air-handling systems are non-safety and non-seismic. Failure of either does not compromise other safety-related air-handling systems or prevent safe shutdown.

The balance of the control complex air-handling system consists of two independent, full capacity systems. Each system serves the associated train of essential electrical equipment areas. Each system is powered from independent Class 1E power sources and served from separate essential chilled water systems. Equipment capacities are based on conservative evaluations of heat-producing equipment and conservative assumptions of surrounding area temperatures. Normally, the electrical equipment areas will be maintained at approximately 85°F. The design basis upper limit of 104°F is based on standard ratings for electrical equipment. J E

The minimum distance between the Control Room Air Intakes and ~~important~~ postulated plant release points is specified on Figure 3.8-5.

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During normal operation and shutdown, the main condenser will have no radioactive contaminants inventory. Radioactive contaminants can only be obtained through primary to secondary system leakage due to steam generator tube leaks. A discussion of the radiological aspects of primary to secondary leakage, including operating concentrations of radioactive contaminants, is included in Chapter 11. There is no hydrogen buildup in the main condenser. A

The main condenser is non-safety-related. Insert (A)

A leak or failure in the condenser shell would allow condensate to drain out, but the pits located below the condenser will hold more water than the condensate hotwell volume. ~~The flooding due to a loss of condenser water box or circulating water piping would be limited to the turbine building which contains no safety-related equipment.~~ I

A failure in the recirculating water system or the main condenser large enough to cause flooding will be detected by high level alarms in the turbine building sumps. The operator can isolate the appropriate equipment.

10.4.1.4 <u>Tests and Inspections</u> A

The main condenser is tested in accordance with the Heat Exchanger Institute Standards for Steam Surface Condensers. The condenser is designed to be capable of being filled with water for hydrotests. The condenser shells, hotwells, and waterboxes are provided with access openings to permit inspection and repairs; periodic visual inspections of and preventive maintenance on condenser components are conducted following normal industrial practice. The condenser and the circulating water system are designed such that isolation of a portion of the tubes to permit repair of leaks is possible. J

10.4.1.5 <u>Instrumentation Applications</u>

All of the instrumentation for this system is operating instrumentation and none is required for safe shutdown of the reactor. Sufficient instrumentation is provided throughout the plant power generation systems to facilitate an accurate heat energy balance on the plant. A

J

Hotwell level and pressure indications are provided locally, and associated alarms are provided in the control room for each condenser shell. The condensate level in the hotwell is

INSERT A

Flooding due to failure of a condenser water box expansion joint or circulating water piping would be limited in the Turbine Building by release of excess water to the yard through openings, doors, and openings caused by shearing of building siding bolts. The ground elevation of the Turbine Building is located above the finished plant grade level to ensure water drainage away from the Turbine Building. All Turbine Building interconnections and miscellaneous pipe tunnels to safety related structures will either be above the maximum internal flood level (associated with the failure of the circulating water piping or condenser water box expansion joint), or sealed to prevent back-flooding. Flooding of safety related plant structures from Turbine Building sources are precluded since the plant grade is sloped away from the safety-related structures for proper drainage. Since the Turbine Building contains no safety related equipment, and no other building is affected by Turbine Building flooding, the impact of internal flooding from the Turbine Building is limited to non-safety related equipment in the Turbine Building.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR TURBINE BUILDING (2.1.2)

1. Amplifying Information for the Turbine Building

Addition to CESSAR-DC Section 10.4.1.3

(See also Nuclear Island Structures, 2.1.1)

Flooding due to failure of a condenser water box expansion joint or circulating water piping would be limited in the Turbine Building by release of excess water to the yard through openings, doors, and openings caused by shearing of building siding bolts. The ground elevation of the Turbine Building is located above the finished plant grade level to ensure water drainage away from the Turbine Building. All Turbine Building interconnections and miscellaneous pipe tunnels to safety related structures will either be above the maximum internal flood level (associated with the failure of the circulating water piping or condenser water box expansion joint), or sealed to prevent back-flooding. Flooding of safety-related plant structures from Turbine Building sources is precluded since the plant grade is sloped away from the safety-related structures for proper drainage. Since the Turbine Building contains no safety-related equipment, and no other building is affected by Turbine Building flooding, the impact of internal flooding from the Turbine Building is limited to non-safety-related equipment in the Turbine Building.

2. CESSAR-DC Chapter 14 Tests Applicable to the TB

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING
(2.1.2)

Relationship of the Safety Analysis to the TB

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING
(2.1.2)

Relationship of the PRA to the TB

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING (2.1.2)

Relationship of the Shutdown Risk Evaluation to the TB

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR THE COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE (2.1.3)

1. Amplifying Information for the CCW Heat Exchanger Structure

   See Nuclear Island Structures (2.1.1) for markups of CESSAR DC.

2. CESSAR-DC Chapter 14 Tests Applicable to the CCW Heat Exchanger Structure

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE (2.1.3)

Relationship of the Safety Analysis to the CCW Heat Exchanger Structure

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE
(2.1.3)

Relationship of the PRA to the CCW Heat Exchanger Structure

None

REFERENCE INFORMATION FOR THE COMPONENT COOLING WATER HEAT EXCHANGER STRUCTURE
(2.1.3)

Relationship of the Shutdown Risk Evaluation to the CCW Heat Exchanger Structure

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR THE DIESEL FUEL STORAGE STRUCTURE (2.1.4)

1. Amplifying Information for the DFSS

   See Nuclear Island Structures (2.1.1) for markups of CESSAR DC.

2. CESSAR-DC Chapter 14 Tests Applicable to the DFSS

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE DIESEL FUEL STORAGE STRUCTURE
(2.1.4)

Relationship of the Safety Analysis to the DFSS

None

REFERENCE INFORMATION FOR THE DIESEL FUEL STORAGE STRUCTURE (2.1.4)

Relationship of the PRA to the DFSS

None

REFERENCE INFORMATION FOR THE DIESEL FUEL STORAGE STRUCTURE
(2.1.4)

Relationship of the Shutdown Risk Evaluation to the DFSS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR THE RADWASTE BUILDING (2.1.5)

1. Amplifying Information for the Radwaste Building

   See Attached

2. CESSAR-DC Chapter 14 Tests Applicable to the Radwaste Building

   None

- foundations and walls of the Radwaste Building are designed to withstand a Safe Shutdown Earthquake (SSE),
- building elements meet the structural requirements specified in AISC N-690 and ACI 349-1985 for steel and concrete respectively,
- curbing or an elevated threshold are provided to prevent the uncontrolled release of liquid or solid effluent to the environment due to a LWMS or SWMS failure,
- floor drains collect and route spills to the LWMS for processing,
- structures housing the Gaseous Waste Management System charcoal absorbers are designed in accordance with Position 5 of Regulatory Guide 1.143 to withstand an SSE.
- tanks internal to the Radwaste Building are designed to a sufficient height to contain the maximum expected inventory in the tank.

B. The Radwaste Building is located adjacent to the Nuclear Annex. Provisions are included to preclude an uncontrolled release of effluents to the environment due to a LWMS or SWMS failure. If site specific requirements preclude locating the Radwaste Building adjacent to the Nuclear Annex, the consequences of a LWMS or SWMS failure are evaluated to demonstrate compliance with 10 CFR 20, Appendix B limits. This analysis demonstrates that the concentration of the liquid effluent at the potable water source, released from the Radwaste Building due to a LWMS or SWMS failure, is within 10 CFR 20, Appendix B limits.

C. The Radwaste Building is located in close proximity to the Interim Onsite Storage Facility to facilitate transport of packaged waste for interim storage prior to shipment to a licensed burial facility.

D. Adequate space is provided for storage and processing of radwaste.

E. Ventilation ensures controlled and monitored release of gaseous effluent from the Radwaste Building.

F. The Radwaste Building is equipped with area and airborne radiation monitors to provide indication of a spill and to ensure that personnel exposures are maintained ALARA.

Insert A

Insert A

(Add paragraph G. to Section 1.2.16.4 for the design criteria for the Radwaste Building as follows:)

G. Pursuant to the requirement in Regulatory Guide 1.143 which requires that the foundations and walls of the Radwaste Building contain the maximum liquid inventory expected in the building, a capacity analysis will be performed which totals the maximum amount of free liquid inventory expected to exist in the as-built piping, tanks, heat exchangers, and material handling equipment of the liquid and solid radioactive waste management systems, and compares that total with the volume-carrying capacity of the portion of the building which is designed to the required seismic criteria in order to contain that total. The requirement shall be considered met if the as-built liquid inventory total is less than the as-built Radwaste Building volume-carrying capacity.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE RADWASTE BUILDING
(2.1.5)

Relationship of the Safety Analysis to the Radwaste Building

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE RADWASTE BUILDING (2.1.5)

Relationship of the PRA to the Radwaste Building

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE RADWASTE BUILDING (2.1.5)

Relationship of the Shutdown Risk Evaluation to the Radwaste Building

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR REACTOR VESSEL INTERNAL STRUCTURES (2.1.6)

1. Amplifying Information for the RVI

   A description of the Comprehensive Vibration Assessment Program (CVAP) is provided in CESSAR-DC Section 3.9.2.4.

2. CESSAR-DC Chapter 14 Tests Applicable to the RVI

   None. Covered by testing description in CESSAR-DC Section 3.9.2.4.

REFERENCE INFORMATION FOR REACTOR VESSEL INTERNAL STRUCTURES (2.1.6)

Relationship of the Safety Analysis to the RVI

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR REACTOR VESSEL INTERNAL STRUCTURES (2.1.6)

Relationship of the PRA to the RVI

None

REFERENCE INFORMATION FOR REACTOR VESSEL INTERNAL STRUCTURES (2.1.6)

Relationship of the Shutdown Risk Evaluation to the RVI

Shutdown Risk instrumentation is identified in the discussion of the Reactor Coolant System.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR CONTROL ELEMENT DRIVE MECHANISM (2.2.4)

1. Amplifying Information for the CEDM

   See CESSAR-DC Section 4.5.1

2. CESSAR-DC Chapter 14 Tests Applicable to the CEDM

   See CESSAR-DC Section 14.2.12.1.37

REFERENCE INFORMATION FOR CONTROL ELEMENT DRIVE MECHANISM (2.2.4)

Relationship of the Safety Analysis to the CEDM

The CEDM releases the CEA upon termination of electrical power to the CEDM.

REFERENCE INFORMATION FOR CONTROL ELEMENT DRIVE MECHANISM (2.2.4)

Relationship of the PRA to the CEDM

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR CONTROL ELEMENT DRIVE MECHANISM (2.2.4)

Relationship of the Shutdown Risk Evaluation to the CEDM

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR NEW FUEL STORAGE RACKS
(2.7.1)

1. Amplifying Information for the NFSR

   The NFSR are discussed in CESSAR-DC Section 9.1.1.

2. CESSAR-DC Chapter 14 Tests Applicable to the NFSR

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR NEW FUEL STORAGE RACKS
(2.7.1)

Relationship of the Safety Analysis to the NFSR

None

REFERENCE INFORMATION FOR NEW FUEL STORAGE RACKS (2.7.1)

Relationship of the PRA to the NFSR

None

REFERENCE INFORMATION FOR NEW FUEL STORAGE RACKS
(2.7.1)

Relationship of the Shutdown Risk Evaluation to the NFSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR SPENT FUEL STORAGE RACKS (2.7.2)

1. Amplifying Information for the Spent Fuel Storage Racks

   The Spent Fuel Storage Racks are discussed in CESSAR-DC Section 9.1.2.

2. CESSAR-DC Chapter 14 Tests Applicable to the SFSR

   None

REFERENCE INFORMATION FOR SPENT FUEL STORAGE RACKS (2.7.2)

Relationship of the Safety Analysis to the SFSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR SPENT FUEL STORAGE RACKS (2.7.2)

Relationship of the PRA to the SFSR

None

REFERENCE INFORMATION FOR SPENT FUEL STORAGE RACKS (2.7.2)

Relationship of the Shutdown Risk Evaluation to the SFSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR POOL COOLING AND PURIFICATION SYSTEM
(2.7.3)

1. Amplifying Information for the PCPS

   PCPS Description: CESSAR-DC Section 9.1.3

2. CESSAR-DC Chapter 14 Tests Applicable to the PCPS

   Test Description: CESSAR-DC Section 14.2.12.1.80

REFERENCE INFORMATION FOR POOL COOLING AND PURIFICATION SYSTEM
(2.7.3)

Relationship of the Safety Analysis to the PCPS

None

REFERENCE INFORMATION FOR POOL COOLING AND PURIFICATION SYSTEM (2.7.3)

Relationship of the PRA to the PCPS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR POOL COOLING AND PURIFICATION SYSTEM
(2.7.3)

Relationship of the Shutdown Risk Evaluation to the PCPS

None

SUPPORTIVE INFORMATION FOR THE CONDENSATE STORAGE SYSTEM
(2.7.8)

1. Amplifying Information for the CSS

   None

2. CESSAR-DC Chapter 14 Tests Applicable to the CSS

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSATE STORAGE SYSTEM (2.7.8)

Relationship of the Safety Analysis to the CSS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSATE STORAGE SYSTEM (2.7.8)

Relationship of the PRA to the CSS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSATE STORAGE SYSTEM (2.7.8)

Relationship of the Shutdown Risk Evaluation to the CSS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR INSTRUMENT AIR SYSTEM
(2.7.10)

1. Amplifying Information for the IAS

   CESSAR DC Section 9.3.1

2. CESSAR-DC Chapter 14 Tests Applicable to the IAS

   See CESSAR-DC Section 14.2.12.1.88

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR INSTRUMENT AIR SYSTEM
(2.7.10)

Relationship of the Safety Analysis to the IAS

Safety systems supplied by IAS will not be rendered inoperable by loss of air supply. [Safety systems incorporate fail-safe configurations for air operated components and equipment.]

REFERENCE INFORMATION FOR INSTRUMENT AIR SYSTEM
(2.7.10)

Relationship of the PRA to the IAS

None

REFERENCE INFORMATION FOR INSTRUMENT AIR SYSTEM (2.7.10)

Relationship of the Shutdown Risk Evaluation to the IAS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR TURBINE BUILDING COOLING WATER SYSTEM (2.7.11)

1. Amplifying Information for the TBCWS

   CESSAR-DC Section 9.2.8

2. CESSAR-DC Chapter 14 Tests Applicable to the TBCWS

   CESSAR-DC Section 14.2.12.1.81

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING COOLING WATER SYSTEM (2.7.11)

Relationship of the Safety Analysis to the TBCWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING COOLING WATER SYSTEM (2.7.11)

Relationship of the PRA to the TBCWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR TURBINE BUILDING COOLING WATER SYSTEM (2.7.11)

Relationship of the Shutdown Risk Evaluation to the TBCWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR THE TURBINE BUILDING SERVICE WATER SYSTEM (2.7.14)

1. Amplifying Information for the TBSWS

   None

2. CESSAR-DC Chapter 14 Tests Applicable to the TBSWS

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE TURBINE BUILDING SERVICE WATER SYSTEM (2.7.14)

Relationship of the Safety Analysis to the TBSWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE TURBINE BUILDING SERVICE WATER SYSTEM (2.7.14)

Relationship of the PRA to the TBSWS

None

REFERENCE INFORMATION FOR THE TURBINE BUILDING SERVICE WATER SYSTEM (2.7.14)

Relationship of the Shutdown Risk Evaluation to the TBSWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR THE COMPRESSED GAS SYSTEM (2.7.27)

1. Amplifying Information for the CGS

   None

2. CESSAR-DC Chapter 14 Tests Applicable to the CGS

   Refer to CESSAR-DC Section 14.2.12.1.89

REFERENCE INFORMATION FOR THE COMPRESSED GAS SYSTEM (2.7.27)

Relationship of the Safety Analysis to the CGS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE COMPRESSED GAS SYSTEM (2.7.27)

Relationship of the PRA to the CGS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE COMPRESSED GAS SYSTEM
(2.7.27)

Relationship of the Shutdown Risk to the CGS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR POTABLE AND SANITARY WATER SYSTEMS
(2.7.28)

1. Amplifying Information for the Potable and Sanitary Water Systems

   None

2. CESSAR-DC Chapter 14 Tests Applicable to the PSWS

   None

REFERENCE INFORMATION FOR POTABLE AND SANITARY WATER SYSTEMS
(2.7.28)

Relationship of the Safety Analysis to the PSWS

None

REFERENCE INFORMATION FOR POTABLE AND SANITARY WATER SYSTEMS (2.7.28)

Relationship of the PRA to the PSWS

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR POTABLE AND SANITARY WATER SYSTEMS (2.7.28)

Relationship of the Shutdown Risk Evaluation to the PSWS

None

SUPPORTIVE INFORMATION FOR THE CONDENSER CIRCULATING WATER SYSTEM (2.8.9)

1. Amplifying Information for the Condenser Circulating Water System

   None

2. CESSAR-DC Chapter 14 Tests Applicable to the Condenser Circulating Water System

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSER CIRCULATING WATER SYSTEM (2.8.9)

Relationship of the Safety Analysis to the Condenser Circulating Water System

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSER CIRCULATING WATER SYSTEM (2.8.9)

Relationship of the PRA to the Condenser Circulating Water System

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR THE CONDENSER CIRCULATING WATER SYSTEM (2.8.9)

Relationship of the Shutdown Risk Evaluation to the Condenser Circulating Water System

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR MAIN CONTROL ROOM (2.12.1)

Amplifying Information for the Main Control Room

1. The following documents submitted on the docket are the bases for the MCR design:

   a. Human Factors Standards Guidelines and Bases (NPX80-IC-DR791-02)

   b. Nuplex 80+ Design Bases (NPX80-IC-DB-790-01)

   c. System Description for Control Complex Information System (NPX80-IC-SD791-01)

   d. System Description for Critical Function and Success Path Monitoring (NPX80-IC-SD790-02)

   e. Functional Task Analysis Methodology (CESSAR-DC Section 18.5)

   f. Operating Experience Review for SYSTEM 80+ MMI Design (NPX80-IC-RR790-01)

   g. Human Factors Program Plan for the SYSTEM 80+ Standard Plant Design (NPX80-IC-DP790-01)

   h. Human Factors Engineering Verification and Validation Plan for NUPLEX 80+ (NPX80-IC-DP790-03)

   i. NUPLEX 80+ Verification Analysis Report (NPX80-TE790-01)

   j. NUPLEX 80+ Function Analysis and Allocation Report

   See CESSAR-DC Section 18.6 for a discussion of the MCR configuration.

2. CESSAR-DC Chapter 14 Tests Applicable to the MCR

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR MAIN CONTROL ROOM (2.12.1)

Relationship of the Safety Analysis to the MCR

Annunciators, Displays, and Controls credited in the Safety Analysis are included in the minimum set of annunciators, displays and controls in the MCR.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR MAIN CONTROL ROOM
(2.12.1)

Relationship of the PRA to the MCR

Annunciators, displays and controls required to execute PRA-significant tasks are included in the minimum set of annunciators, displays and controls in the MCR.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR MAIN CONTROL ROOM (2.12.1)

Relationship of the Shutdown Risk Evaluation to MCR

Annunciators, displays and controls related to shutdown risk are included in the minimum set of annunciators, displays and controls in the MCR.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR REMOTE SHUTDOWN ROOM (2.12.2)

1. Amplifying Information for the RSR

   See the "Amplifying Information" for ITAAC 2.12.1 for documents which are the bases for the RSR design.

   See CESSAR-DC Section 18.8 for a discussion of the RSR.

2. CESSAR-DC Chapter 14 Tests Applicable to the RSR

   None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR REMOTE SHUTDOWN ROOM
(2.12.2)

Relationship of the Safety Analysis to the RSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR REMOTE SHUTDOWN ROOM (2.12.2)

Relationship of the PRA to the RSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR REMOTE SHUTDOWN ROOM (2.12.2)

Relationship of the Shutdown Risk Evaluation to the RSR

None

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

1. Amplifying Information for the Control Panels

See Section 2.12.1 amplifying information for documents which are the bases for the Control Panel design.

The following applications are standardized Human-System Interface (HSI) features that utilize consistent operating conventions at Nuplex 80+ control panels:

DPS Display Hierarchy
DIAS Alarm Tile Displays
DIAS Dedicated Parameter Displays
DIAS Multiple Parameter Displays
CCS Process Controller Displays
CCS Switch Configurations

STANDARD FEATURE: DPS Display Hierarchy

The DPS Display Hierarchy is a standard Human-System Interface (HSI) feature of the Nuplex 80+ Data Processing System (DPS). The DPS Display Hierarchy submitted for final design approval is described in CESSAR-DC (Chapter 18). The major characteristics of the DPS Display Hierarchy are as follows:

1. The DPS Display Hierarchy is an integrated presentation of Nuplex 80+ process information.

2. The DPS Display Hierarchy provides access to displays and annunciators for system status and process parameters.

3. Touch-screen VDU devices are utilized.

4. On each display page in the DPS Display Hierarchy, a spatially dedicated message area and main menu are provided.

5. The DPS Display Hierarchy permits selectable access to any of its display pages from any DPS terminal.

6. The DPS Display Hierarchy permits acknowledgment of Nuplex 80+ annunciators.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

7. The DPS Display Hierarchy automatically provides specific alarm condition messages at the time of alarm acknowledgment.

8. The DPS Display Hierarchy is configured to conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

9. The DPS Display Hierarchy indications are read at the panel.

10. The DPS VDU devices are located on the vertical panel sections.

11. The DPS Display System is diverse and independent of the Discrete Indication and Alarm System (DIAS).

STANDARD FEATURE: DIAS Alarm Tile Display

The DIAS Alarm Tile Display is a standard Human-System Interface (HSI) feature of the Nuplex 80+ Discrete Indication and Alarm System (DIAS). The DIAS Alarm Tile Display submitted for final design approval is described in CESSAR-DC (Chapter 18). The major characteristics of the DIAS Alarm Tile Displays are as follows:

1. Software-generated alarm tiles present groups of functionally-related alarm status messages.

2. Touch-screen VDU devices are utilized.

3. On each DIAS Alarm Tile Display device, the status of alarm tiles is presented on a single alarm tile display page; for each tile, an associated alarm list page is available to present the status of the individual alarm conditions.

4. Unacknowledged alarms on a single tile are acknowledged through the display as a group.

5. Alarm condition messages are automatically provided upon alarm tile acknowledgment.

6. Alarm tiles are assigned to control panels by corresponding plant systems.

7. Alarm tile display devices are located on the vertical panel sections.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

8. Alarm tiles on the alarm tile display page are spatially dedicated.

9. DIAS Alarm Tile Displays are configured to conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

10. Tile details are read at its panel; tile status is visible across the controlling work space.

11. Alarm tiles are established for process parameters that provide direct indication of:

    a. Critical Safety Functions
    b. Critical Power Production Functions
    c. Success Path performance
    d. Success Path availability
    e. Damage to major equipment
    f. Personnel hazard

12. Alarms are presented in one of four states: new, existing, cleared, reset.

13. The highest priority of new or cleared alarm state, and the highest priority of existing alarm state are both provided by each individual tile.

14. A tile "stop/resume flash" feature is provided for Priority 2 and 3 alarms.

15. A momentary tone provides an initial audible alert of the transition of one or more alarms to new or cleared states for priority 1 or 2 alarms.

16. A momentary reminder tone provides a recurring audible alert if Priority 1 or 2 alarms remain unacknowledged.

17. Alarm tones emit from the console where the alarming display is located.

STANDARD FEATURE: DIAS Dedicated Parameter Display

The DIAS Dedicated Parameter Display is a standard Human-System Interface (HSI) feature of the Nuplex 80+ Discrete Indication and Alarm System (DIAS). The DIAS Dedicated Parameter Display submitted for final design approval is described in CESSAR-DC (Chapter 18). The major characteristics of the DIAS Dedicated Parameter Displays are as follows:

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

1. DIAS Dedicated Parameter Displays are software-generated display representations of process parameters. Each dedicated parameter display presents a single value based on redundant sensor data.

2. DIAS Dedicated Parameter Displays present validated information based on redundant sensor data. Validation failures are indicated on the displays.

3. DIAS Dedicated Parameter Displays present spatially dedicated information.

4. A DIAS Dedicated Parameter Display permits continuous display of the individual data points.

5. DIAS Dedicated Parameter Displays incorporate automatic range change features.

6. Touch-screen VDU devices are utilized.

7. DIAS Dedicated Parameter Displays are assigned to control panels by corresponding plant systems.

8. DIAS Dedicated Parameter Display devices are located on the vertical control panel sections.

9. DIAS Dedicated Parameter Displays are configured to conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

10. DIAS Dedicated Parameter Display values are read from across the Main Control Console; the Display details are read at the panel.

11. DIAS Dedicated Parameter Displays are provided for the following:

    a. Critical Safety Functions
    b. Success Path performance
    c. PAMI indication
    d. Reg. Guide 1.97

12. DIAS Dedicated Parameter Displays are diverse and independent of the DPS display system.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

STANDARD FEATURE: DIAS Multiple Parameter Display

The DIAS Multiple Parameter Display is a standard Human-System Interface (HSI)feature of the Nuplex 80+ Discrete Indication and Alarm System (DIAS). The DIAS Multiple Parameter Display submitted for final design approval is described in CESSAR-DC (Chapter 18). The major characteristics of the DIAS Multiple Parameter Displays are as follows:

1. DIAS Multiple Parameter Displays are software-generated display representations of process parameters.

2. DIAS Multiple Parameter Displays present validated information based on redundant sensor data. Validation failures are indicated on the displays.

3. DIAS Multiple Parameter Displays are digital and analog representations of process parameters.

4. A DIAS Multiple Parameter Display permits continuous display of any one of its individual data points.

5. Touch-screen VDU devices are utilized.

6. Multiple parameters are assigned to control panels and combined into common DIAS Multiple Parameter Display devices based on plant systems relationships.

7. DIAS Multiple Parameter Display devices are located on the vertical control panel sections.

8. DIAS Multiple Parameter Displays are configured to conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

9. DIAS Multiple Parameter Display values are read at the panel.

10. DIAS Multiple Parameter Displays are diverse and independent of the DPS display system.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

SUPPORTIVE INFORMATION FOR CONTROL PANELS
(2.12.3)

STANDARD FEATURE: CCS Process Controller Display

The CCS Process Controller Display is a standard Human-System Interface (HSI) feature of the Nuplex 80+ Component Control Systems (CCS). The CCS Process Controller Display submitted for final design approval is described in CESSAR-DC (Chapter 18). The major characteristics of the CCS Process Controller Displays are as follows:

1. CCS Process Controller Displays are software-generated representations of process control devices and their controlled variables.

2. Touch-screen VDU devices are utilized.

3. CCS Process Controller Display devices are located on the control panel bench board sections.

4. CCS Process Controller Displays conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

5. CCS Process Controller Displays are read at the panel.

6. Controls are assigned to control panels based on plant systems, and are combined into Process Controller Display devices based on shared functional relationships.

7. CCS Process Controller Display is divided into sections for master loop and sub loop control and monitoring.

8. CCS Process Controller Displays permits selection of operating modes, loop control signal, and loop setpoints.

9. CCS Process Controller is a man-machine interface device only. All control loop electronics are located outside the main control room.

STANDARD FEATURE: CCS Switch Configuration

The CCS Switch Configurations are a standard Human-System Interface (HSI) feature of the Nuplex 80+ Component Control Systems (CCS). The CCS Switch Configurations submitted for final design approval are described in CESSAR-DC

(Chapter 18). The major characteristics of the CCS Switch Configurations are as follows:

1. CCS Switch Configurations utilize physical push buttons with backlit legend status indicators.

2. CCS Switch Configurations permit on-line replacement and bumpless transfer.

3. CCS Switch Configurations are assigned to control panels based on plant systems, and combined into multiple component units based on functional relationships.

4. CCS Switch Configuration devices are located on the control panel bench board sections.

5. CCS Switch Configurations conform to the System 80+ Human Factors Standards, Guidelines, & Bases.

6. CCS Switch Configuration details are read at the panel.

See CESSAR-DC Section 18.7 for discussion of information presentation and panel layout.

2. CESSAR-DC Chapter 14 Tests Applicable to the Control Panels

None

REFERENCE INFORMATION FOR CONTROL PANELS
(2.12.3)

Relationship of the Safety Analysis to the Control Panels

Annunciators, displays, and controls credited in the Safety Analysis are included in the minimum set of annunciators, displays and controls in the MCR.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR CONTROL PANELS
(2.12.3)

Relationship of the PRA to the Control Panels

Annunciators, displays, and controls credited in the PRA are included in the minimum set of annunciators, displays and controls in the MCR.

For reference purposes only. Not intended to comprise a part of either the Tier 1 or Tier 2 System 80+ submittal.

REFERENCE INFORMATION FOR CONTROL PANELS
(2.12.3)

Relationship of the Shutdown Risk Evaluation to the Control Panels

Annunciators, displays, and controls credited in the Shutdown Risk Report are included in the minimum set of annunciators, displays and controls in the MCR.